Session 4

Radiation protection of patients, staff and the public in diagnostic and therapeutic nuclear medicine and hybrid imaging

AN ANALYSIS ON STAFF EYE LENS DOSES OF SELECTED OCCUPATIONAL CATEGORIES AT A LARGE MEDICAL CENTER IN SAUDI ARABIA

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Abstract

A 5-year retrospective analysis of the eye doses was performed on staff from different occupational categories namely cyclotron, nuclear medicine and PET/CT at King Faisal Specialist Hospital and Research Centre (KFSHRC) in Riyadh, Saudi Arabia. KFSHRC is a tertiary medical centre with 1,000-bed with the largest cyclotron facility in the Gulf region. The aim of the study is to assess staff doses to the lens of the eyes using values from the annual TLD dose report for years 2012 to 2016 and determine the category of staff with high estimated eye doses. The study also aims to investigate the causes for high doses and recommend dose-reduction techniques. The dose to the lens of the eye was estimated by using the ratio Hp(0.07)slab/Hlens equals 1.1 where Hp(0.07) values are the reported doses from TLD badge worn at the collar level. The average annual eye dose of each category for the 5-y monitoring period was determined and compared with the eye lens doses of cardiologists in the previous study. Radation workers in cyclotron facilities obtained the highest eye lens dose compared to nuclear medicine and PET/CT doses.

1. INTRODUCTION

The International Atomic Energy Agency (IAEA) in its new basic safety standards [1] has changed the annual eye lens dose limit for radiation workers from 150 mSv to 20 mSv averaged over five years but not to exceed 50 mSv in a single year. The new eye lens dose limit is based on epidemiological studies considered in the new recommendations of the International Commission on Radiological Protection (ICRP) issued in 2007 [2-3]. IAEA states that the category of workers who are likely to exceed the new eye lens dose limit are those who work in interventional radiology, interventional cardiology, PET/CT radiopharmaceutical preparation and cyclotron facilities [4]. The study of Sensakovic et al [5] showed that about 1% of the monitored radiation workers can exceed the new limit.

Occupational workers working in interventional radiology and cardiology have higher possibility of exceeding the new eye lens dose limit. The ORAMED study showed that radiologists and cardiologists measured eye lens doses ranged fron 10 µSv to 4 mSv per procedure and about 24% exceeded the eye lens dose limit [6]. The study of Omar et al showed that the primary cardiologists have 66 µSv (third quartile) per procedure and 300 procedures should be performed annually so as not to exceed the eye lens dose limit of 20 mSv per year [7] and the study of Kim et al showed values of eye lens doses could range from 0.4 to 1,100 µSv per procedure depending on the use of protective shields [8]. The RAD-IR study showed that the estimated eye lens doses of interventionists could reach a maximum value of 11 mSv per procedure [9]. In Saudi Arabia , cardiologists could have an average annual eye lens dose of 2 mSv with a maximum of 5 mSv and could possibly exceed the limit if workload is increased [10-11]. In Nuclear Medicine, the staff who perform injection and scanning in PET/CT are the most exposed to high whole body doses due to the high energy of radioactive Fluorine. The eye lens dose of the technologists could be twice the whole body dose. Studies show that PET/CT staff who prepares and administers F-18 (FDG) can have a whole body dose of about 2 to 6 µSv/procedure and with increased workload, the eye lens dose could approach the new limit [12-15].

The whole body doses Hp(10) and Hp(0.07) can be used to assess the eye lens. Using a calibrated and type tested whole body dosimeter in a standard ICRU slab phantom the Hp(10) and Hp(0.07) measurements can be used as a surrogate for eye lens dose measurements[4, 16].

King Faisal Specialist Hospital & Research Centre in Riyadh, Saudi Arabia is the largest medical institution with the biggest cyclotron facility in the Middle East. In our 5 year study from 1995 to 1999, monitoring of occupational doses of radiation workers in cyclotron, cardiac catheterization laboratory and nuclear medicine facilities ranked to be the categories with high measurable values of whole body doses [17]. The study aims to assess the eye lens doses of workers in selected occupational categories using the H(0.07)

values and investigate factors that could increase the possibility of approaching or exceeding the new eye lens dose limit.

2. METHODS

All radiation workers at KFSHRC are monitored for Hp(10) and Hp (0.07) using the TLD 100 dosimeters. The TLD dosimeters used are type-tested at its Secondary Standard Dosimetry Laboratory (SSDL) and calibrated using the standard ICRU slab phantom with measurements of 30 x 30 x 15 cm. Each staff is provided with one TLD badge to be worn at the collar level outside the lead protective shield. The system of dose monitoring at KFSHRC fits the requirement for eye lens dose assessment using either the Hp(10) or the Hp(0.07) values.

A total of 359 records of monitored radiation workers from three occupational categories namely cyclotron, nuclear medicine and PET/CT were included in the study. The monthly dose records of all radiation workers for the five year period covering the years 2012 to 2016 were retrieved. The eye lens dose per staff was estimated using the equation: $Hp(0.07)_{slab}=H_{lens}$ / 1.1 where H_{lens} corresponds to the calculated equivalent dose for the eye lens and $Hp(0.07)_{slab}$ is the calculated personal dose equivalent taken from the readings of TLD badge. The TLD dosimeters were calibrated and type tested using the standard slab phantom representing the personal dose equivalent. Pooling all annual eye H_{lens} values, they were grouped to dose ranges of <0.1, 0.1 to 1, >1 to <5 and 5 mSv and above. A graphic distribution of the dose ranges was made. The category of staff with high eye lens doses was identified. The average eye lens dose was also calculated for each staff category for the 5-y period and compared with the eye lens dose of cardiologists in the same center. Factors that contributed to high eye lens doses were investigated.

3. RESU LTS

The total number of monitored staff for each category from year 2012 to 2016 is presented in Table 1. Cyclotron has the highest number of staff which is about 50% of the total staff in the study. PET/CT has the lowest number of staff which is in the range of 8 to 15 % of the total number of staff.

TABLE 1. NUMBER OF MONITORED RADIATION WORKERS FOR EACH OCCUPATIONAL CATEGORY FROM 2012 TO 2016

Category	2012	2013	2014	2015	2016
Cyclotron	45	50	50	47	43
Nuclear Medicine	15	20	14	18	17
PET/CT	5	6	18	10	11

The distribution of staff for Hp(10) and Hp(0.07) is shown in Fig. 1. PET/CT radiation workers obtained the highest values for Hp(10) and Hp(0.07). The Hp(0.07) values of PET/CT radiation workers are extremely high in 2013 and 2014.

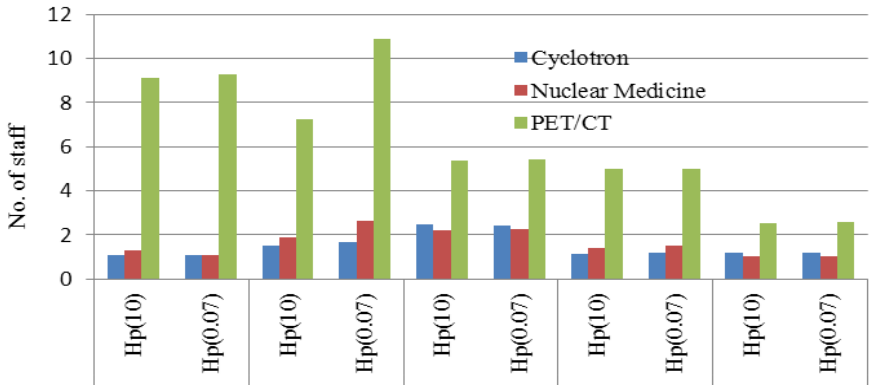

FIG.1. Distribution of monitored radiation workers for Hp(10) and Hp(0.07) from year 2012 to 2016

Pooling all the eye lens doses, the distribution of staff for the dose ranges <0.1, 0.1 to 1, >1 to <5 and 5 mSv and above is shown in Fig. 2. The graph shows that the annual eye lens dose values can be greater than 5 mSv with the highest number of staff in year 2013 and 2014. There was a large decrease in number of staff for the range of 5 mSv and above in year 2015 and no staff was in this range in 2016.

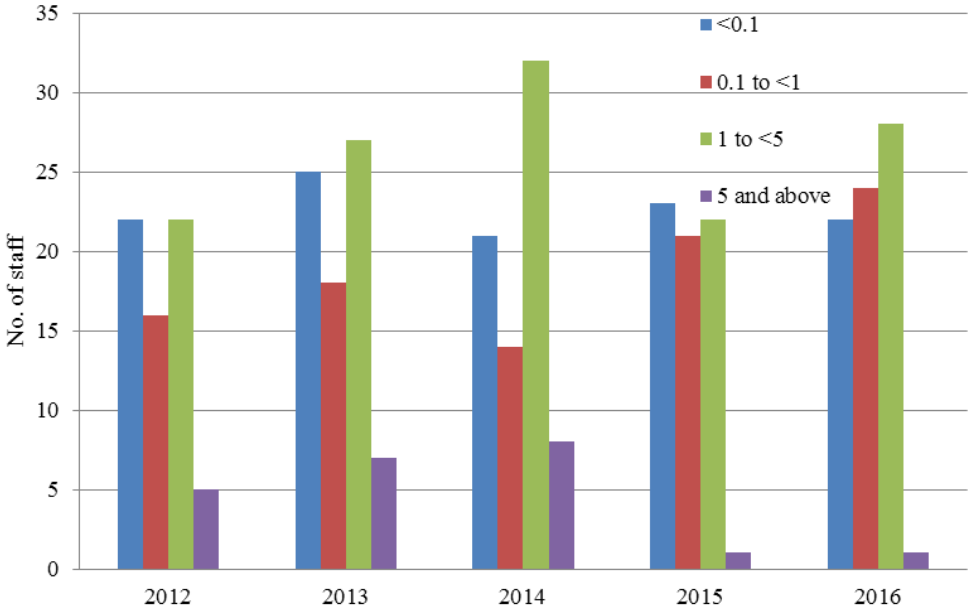

FIG. 2. Distribution of staff for each dose range of eye lens doses for year 2012 to 2016.

The calculated average annual eye lens doses for each category of workers are listed in Table 2. The PET/CT monitored staffs have wide variation in the annual eye lens doses from 2012 to 2014 and decreased in 2015 and 2016.

TABLE 2. AVERAGE ANNUAL EYE LENS DOSE AND STANDARD DEVIATION FOR EACH CATEGORY FROM 2012 TO 2016

Category	2012	2013	2014	2015	2016
Cyclotron	1.2±2.0	1.55±2.2	2.21±2.4	1.1±1.1	1.1±1.3
Nuclear Medicine	1.2±1.1	1.4±2.6	2.2±2.8	1.5±1.6	1.35±1.1
PET/CT	10.2±4.6	9.9±5.8	4.9±4.5	4.57±3.9	2.56±2.1

4. DISCUSSIONS

The number of staff working in the categories varies every year due to expansion of services. The cyclotron department has one very old unit and the number of cyclotron units increased in the year 2013-2014. The number of monitored staff per year varies due to staff resignation. PET/CT has the lowest number of staff in 2012 because PET/CT was newly installed. The number of staff started to increase in 2014 due to the additional PET/CT unit.

The average annual eye lens doses of cyclotron monitored staff are in the range of 1mSv. The increase in the eye lens dose in year 2014 is due to the major repair made on the big cyclotron for a long period of time. For nuclear medicine, the eye lens dose increased by a factor of 2 in year 2014. This is due to the sudden decrease in the number of staff (Table1) with a yearly increasing workload.

PET/CT monitored staff obtained the highest eye lens doses in year 2012 and 2013. During these two years, PET/CT procedures were introduced using the manual injection. The decrease in the dose by half in 2015

and 2016 is due to the introduction of the automatic injector and the implementation of the radiation protection program. It is recommended that staff be rotated at different phases of the procedure from administration to imaging during manual injection of the radiopharmaceutical [14].

5. CONCLUSIONS

Cyclotron staff will have an increased eye lens dose during equipment repair. Although their eye lens doses will increase during this period, there is a remote possibility that the new lens dose limit will be exceeded. However, the complexity of the problem during repair will vary. It will benefit the staff if the eye lens doses be monitored during the repair period. Staff working in the PET/CT facility will be at risk of exceeding the new eye lens dose limit if manual injector is used and if there is a lack of proper radiation safety training. Implementation of the radiation protection program is an important step to optimize staff protection.

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Practical Internal Dosimetry in Radioactive Iodine-131 therapy in a Newly Established Hospital: Challenges and Obstacles

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Abstract

Purpose: This study is aimed to find out the absorbed dose delivered to blood in patients with Carcinoma thyroid who received fixed doses of Radioactive Iodine (RAI)-131 in accordance with different clinical scenarios.

Methods and Materials: Sixty six patients were included in this study. The administered activity ranged from 80-300 mCi and was given orally to all patients. The dose rate (µSv/h)at 1 m from the patients was measured at 24 hourly intervals with survey meter. The whole body residence time was calculated from a graph plotted from the time of reading (hour) and dose rate (µSv/h) on a semi log paper. The Total Blood Volume (TBV) was calculated and finally the absorbed dose to the blood was calculated for all patients.

Results: Out of all patients with female to male ratio 2:1, only four patients (2 females & 2 males) the absorbed dose to blood exceeded the threshold value. All these four patients received activity from 201 mCi to 300 mCi, while for the activity from 80 to 200 mCi the dose to the blood was within permissible range. The mean of the residence time in the total body was 28.33 h (14, 94), mean absorbed dose to the blood in mGy/MBq was 0.147101 (0.065386, 0.327427) and mean absorbed dose to the blood in mGy was 905.55 (241.94, 3634.44).

Conclusion: It was concluded that the pre-therapy dosimetry is important for every carcinoma thyroid patient, especially for patients whose administered activity is greater than 200 mCi.

Introduction

Thyroid cancer is a unique group of tumors with very good survival [1, 2]. Its management has changed over the past several decades due to our better understanding of the disease. The management of thyroid cancer is dependent on the assistance of the group of health care providers with diverse expertises [2]. The incidence of the thyroid carcinoma is almost 1-2% of all human cancer [1, 3]. It is estimated that

more than 150, 000 cases of thyroid cancer are diagnosed every year in world with only 6.5% of deaths due to the disease itself. Still Differentiated Thyroid Carcinoma (DTC) is relatively uncommon and accounts for 1.18 peoples per 100,000 persons in world. The incidence of thyroid cancer in USA is about 1.6%. It affected all age groups, but women are thrice more affected than men [2].

The initial management after surgery is RAI-131 therapy [2, 4, 5]. The administered activity for the ablation of remnants and treatment of metastasis is still a matter of debate [6, 7], after the first RAI-131 use by Seidlin et al. in 1946 [6]. The dosimetry concept for maximum administered dose (administered activity) was first introduced by Benua et al. in 1962, who perceived that there is a chance of differentiation after repeated small therapeutic doses of RAI-131 and a chance that tumors may lose iodine concentration capability [6, 8]. Therefore Nuclear Physicians prefer to give initially high doses of RAI-131. The main limitation associated with the high dose/activity for the management of DTC is bone marrow suppression. Because the maximum RAI dose (administered activity) should not be deliver the absorbed dose $\geq$ 2 Gy (200 rad) to the blood, which is equivalent to the whole body retention of 4.4 GBq (120 mCi) at 48 h [3, 6, 8, 9].

RAI-131 activities given to patients are usually empirically fixed [10-12], based on patient's age and disease state. The optimum RAI-131 activity (1-5 GBq) of a single absorbed dose is enough for the ablation of the post-surgical residues, but these values are dissimilar in different centres (1.11, 1.85, 3.7 GBq). According to different guidelines the RAI-131 activity for paediatric patients is also different. They consider different factors such as weight, surface area and age etc. in such calculation [10]. However the disadvantages of empirically fixed dose method are that some patients may be undertreated and in some other the dose to the bone marrow may exceed the limits.

There has been a continuous improvement in the field of dosimetry over the previous years, some societies use dosimetry techniques based on the blood and urine sample measurements, other use the image based on whole body dose determinations. Some other use Medical Internal Radiation Dose (MIRD) technique, patient specific Monte Carlo simulations and dose point kernel convolution dosimetry etc. [6]. Similarly some other calculate RAI activity value from 24 h percentage uptake [10, 13]. Whatever technique a Nuclear Physician/Medical Physicist use, but in management of thyroid cancer by RAI-131 the absorbed dose to the bone marrow as a critical organ cannot be calculated directly, however the concentration of RAI-131 is almost similar in blood, bone marrow and most organs; consequently the dose to the blood is a good approximation of the total exposure from the administered activity to the patients [8]. Therefore the purpose of the study was to find the radiation dose delivered to the blood in empirically fixed radioactive iodine therapy (RAIT) to patients who received RAI-131 therapy at SINOR cancer hospital Saidu Sharif Swat Pakistan.

Methods and Patients

Patients Selection

This prospective study was designed for the patients who were treated with RAI-131 ablation dose in Nuclear Medicine Department. All the patients were admitted at Swat Institute of Nuclear Medicine, Oncology and radiotherapy (SINOR) cancer hospital, Saidu Sharif Swat, Pakistan between January 2015 and December 2016. Sixty six patients with female to male ratio of approximately 2:1 (Female: 46 and Male: 20) with thyroid carcinoma were selected from different rural and urban area for calculation of absorbed dose to the blood. The total patients were divided in two groups (group A &

group B). In group A there were 55 patients for whom the administered activity was less than or equal to 200 mCi. In group B there were 11 patients for whom the administered activity was greater than 200 mCi.

Administered Activity Calculation

The fixed RAI-131 administered activity was decided by the Nuclear Physician with accordance to different clinical scenarios ranged from 80-300 mCi and was given orally to all patients on empty stomach. The activities/doses were calibrated with CAPINTEC Inc.; dose calibrator (Model, CRC^{R}-15R).

Calculation of Residence Time

The dose rate (µSv/h) at 1m from the patients was measured at 24 hourly intervals with survey meter (Model No: RM 1001 RD). The whole body residence time was calculated from a graph plotted from the time of reading (hour) and dose rate (µSv/h) on a semi log paper by using the following formula (1) [14].

$$\tau = \frac{\log_e(1)}{m} \text{----------} (1)$$

Where m; is the slope taken from the graph between log of dose rate and time as shown in the following figure 1.

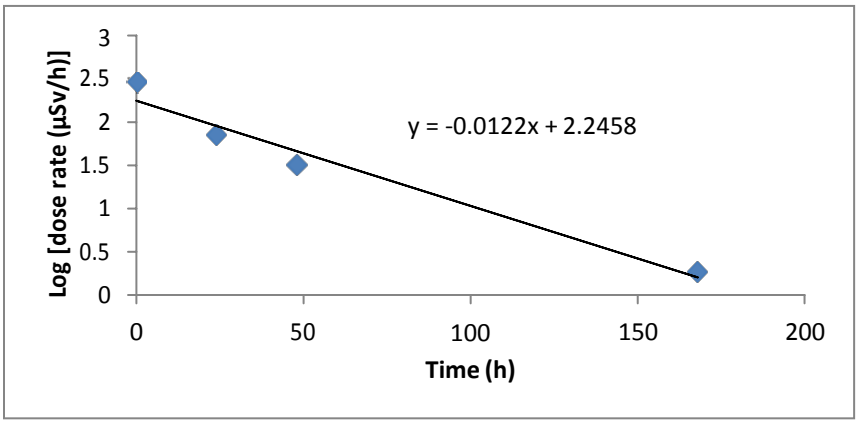

Fig. 1: Graph between log of dose rate (µSv/h) and time of reading (h).

Calculation of Total Blood Volume and Surface Area

The total blood volume (TBV) for each carcinoma thyroid patient was calculated by the following formulas for male and female respectively [15], as shown in the equation (2) and (3).

$$TBV\ (ml) = [(1.06 * age) + (822 * S)] + (1395 * S) \text{ ----------} (2)$$

&

$$TBV(ml) = [(1486 * S) - 825] + (1578 * S) \text{ ----------} (3)$$

Where age is in years and "S" represents surface area in m^2, which was calculated by the formula [15], as demonstrated in equation (4).

$$S = [Wt(kg)]^{.425} * [h(cm)]^{.725} * 0.007184 \text{ ----------} (4)$$

Absorbed Dose to the Blood

The absorbed dose to the to the blood per unit administered activity in mGy/MBq for each patient was calculated by the formula (5) [8].

$$\frac{D_{blood}\,(mGy)}{Ao(MBq)} = \left[\frac{15.12}{BLV(ml)} + \frac{0.0188}{Wt\,(kg)^{\frac{2}{3}}}\right] * \tau^{total\ body} \qquad \text{-------- (5)}$$

Finally the total dose to the blood in mGy for all patients were calculated by using the following equation (6).

$$D\,(mGy) = \frac{D_{blood}\,(mGy)}{Ao\,(MBq)} * Adm.Dose\,(MBq) \qquad \text{---------- (6)}$$

Results and Discussion

The mean values of residence time for the total body (T_{TB}) in hours, dose to the blood per unit administered activity (mGy/MBq) and absorbed dose to the blood in Gy with minimum and maximum values of all patients are presented in the following table (1), for group A and group B respectively.

Table1: The mean values of residence time for the total body (T_{TB}) in hours, dose to the blood per unit administered activity (mGy/MBq) and absorbed dose to blood in Gy with minimum and maximum values of all patients.

Group A (80-200 mCi)				**Group B (201-300 mCi)**			
No. of Patients	T_{TB}	D_{blood} (mGy)/A_o (MBq)	D (Gy)	No. of Patients	T_{TB}	D_{blood} (mGy)/A_o (MBq)	D (Gy)
55	27 (14, 94)	0.14 (0.07,0.32)	0.7 (0.2, 1.8)	11	37 (21, 63)	0.19 (0.09,0.33)	1.9 (1, 3.6)
No. of Patients exceeded threshold dose to the blood			Nil	No. of patients exceeded threshold dose to the blood			4 (2 M & 2F)

From the above data it was noted that only four patients (2 males & 2 females) the absorbed dose to blood exceeded the threshold value. All these four patients received activity from 201 mCi to 300 mCi (in group B), while for the activity from 80 to 200 mCi the absorbed dose to the blood was within permissible range (group A).

The most important variable in the absorbed dose to the blood was total body residence time and even with higher administered activity (201-300 mCi) the limits were not exceeded.

The mean of the residence time for the total body was 28.33 h with minimum and maximum value (14, 94), mean absorbed dose to the blood in mGy/MBq was 0.147101 with minimum and maximum value (0.065386, 0.327427) and mean absorbed dose to the Bone Marrow (BM) in mGy was 905.55 with minimum and maximum value (241.94,3634.44).

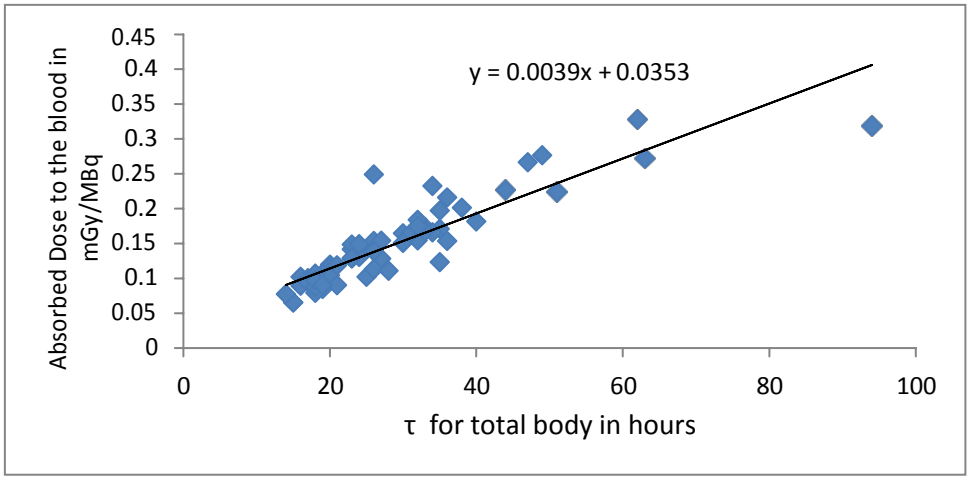

Fig. 2: Graph between residence time for total body (h) and absorbed dose to the blood (mGy/MBq).

The above graph shows that there was a direct relation between residence time (h) for the total body and absorbed dose the blood (mGy/MBq) by the following relation (7).

$$\left(\frac{\text{mGy}}{\text{MBq}}\right) = 0.003 * \tau\ (h) + 0.0353 \qquad (7)$$

The above equation (7) is a very useful equation for the calculation of absorbed dose to the blood per unit administered activity from whole body residence time and this equation will be very useful in future after collecting large data of patients, on the basis of whom the equation will be more accurate and precise.

Factors Affecting Absorbed dose to the Blood

It was observed that the absorbed dose to the blood depends on age, weight, Total Blood Volume (TBV), residence time, surface area, height and administered activity to the patients. The absorbed dose to the blood was greater for those patients having greater residence time, small TBV, small weight, small surface area and less height. The absorbed dose to the blood was smaller for those patients having lesser residence time, major TBV, greater weight, more surface area and more height.

Conclusion

It is concluded that the empirically fixed RAI-131 activity delivered to the patients is safe up to varied extent, but in some patients [4/66 (6.06%) in our total data and 4/11 (36.63%) in patients received activity greater than 200 mCi] the absorbed dose to the blood are greater than the threshold limit. Secondly a very low absorbed dose to blood is not too good, because the target may receive lower absorbed dose than required due to the small radioiodine uptake. So pre-therapy dosimetry is important for every carcinoma thyroid patient, it is especially for those patients whose administered activity is greater than 200 mCi.

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MOVING FROM WEIGHT-ADJUSTED 18F-FDG DOSE TO BODY MASS INDEX (BMI) ADJUSTED DOSE IN RELATION TO RADIATION PROTECTION

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Abstract

Radiation protection awareness and patient radiation dose has been lately the focused topic in PET/CT. Several studies proposed switching from a weight-adjusted 18Fluorine-fluorodeoxyglucose (FDG) dose to BMI-adjusted dose. This potentially reduces patient radiation dose while maintaining image quality. **Method:** Using a PET/CT scanner with a TOF system, thirty-one oncology patients (age 25-76 y, weight 43-114 kg) underwent a PET/CT scan with FDG-weight adjusted dose. A follow up scan for the same patients was evaluated with FDG BMI-adjusted dose. Protocols of all scans were acquired as per IAEA guidelines for PET/CT. Images were evaluated qualitatively by two experienced nuclear medicine physicians using a scoring system and quantitatively by comparing the max standard uptake value (SUV max). **Results:** The injected dose was reduced by 26.9% ± 15.3% with BMI-adjusted dose compared to weight-adjusted dose of the same group of patients. Visual inspection of PET images is the main tool for image interpretation. Good image quality was obtained despite the reduction in the injected dose. SUV max values obtained from both dosing methods were comparable. **Conclusion:** The study showed there was no compromise in image quality with reduction in patient injected dose. This will reduce staff radiation dose and cost as well.

1. INTRODUCTION

The number of whole body 18F-fluorodeoxyglucose (FDG) positron emission tomography/ computed tomography (PET/CT) studies has increased dramatically since the last decade for oncology diagnosis, staging and monitoring response to therapy. Positron emitter 18F-FDG is a widely known radiopharmaceutical. It is produced by the bombardment from the cyclotron with 240 Kev and 511 Kev of beta and gamma energies, respectively [1]. Hence, optimal dosing of 18F-FDG has lately been the focused topic in PET/CT in order to balance exam benefits with the risk from radiation exposure.

In PET/CT imaging, patients are exposed to dual radiation generated from both CT and 18F-FDG. While maintaining diagnostic information, studies considered minimizing patient effective radiation dose received from 18F-FDG as low as reasonably achievable (ALARA), the second general principle of radiological protection as recommended by the international commission of radiological protection (ICRP) [2]. This approach was due to the increased awareness on the risk of ionizing radiation to patients [1,2].

Current procedures in many centers around the world propose switching from the traditional way of only considering patient's weight when prescribing 18F-FDG dose to patients body mass index (BMI) adjusted dose, where both patient's weight and height are taken into account. With this method of dose calculation, the injected 18F-FDG BMI- adjusted dose is reduced, which in return reduces radiation exposure to technologists. Hence, promotes personnel radiation protection and dose savings [3,4,5,6]. This contributes as well in cost reduction and economical savings. The aim of this study was to compare changes in PET/CT image quality as a result of switching from weight to BMI adjusted 18F-FDG dose.

2. MATERIALS AND METHODS

This is a retrospective study observed from April to June 2017 where thirty-one oncology patients (11 men, 20 women; age range 25-76y; mean 49.5± 14.4 y) underwent a whole-body PET/CT scan at least twice at Sultan Qaboos University Hospital.

The weight (43-114 kg; mean 67.56± 16.8) and height of all patients were recorded by the technologists. BMI, weight in kilogram divided by the square of height in meters, of patients (16.94- 36.41 kg/m^2; mean 25.79 ±5.85) was calculated and patients were categorized according to the World Health Organization guidelines into underweight patients (BMI ≤ 18.49 kg/m^2), normal weight patients (BMI 18.5- 24.9 kg/m^2), overweight patients (BMI 25 – 29.9 kg/m^2), obese patients (BMI 30 – 39.9 kg/m^2), and morbidly obese patients (BMI ≥ 40 kg/m^2).

The inclusion criterion of this study were adult patients who underwent a PET/CT scan with BMI adjusted 18F-FDG dosing, as a follow up scan of weight-adjusted 18F-FDG dosing (scan interval 43 - 624 days). The exclusion criteria were patients with liver disease. As higher glucose level results in a lower SUV [7], this study as well excluded patients with glucose levels higher than 11.1 mmol/L.

All patients were instructed to refrain from any strenuous activities for 24 hours prior to the study and fast for at least 6 hours before 18F-FDG administration. A bolus weight adjusted 18F-FDG injection of 6.10-12.11 mCi was administered intravenously into all patients and 10.22 MBq/BMI for underweight patients, 9.30 MBq/BMI for normal-weight patients, 8.18 MBq/BMI for overweight patient and, 7.16 MBq/BMI for obese patients was administered to same group of patients at the follow up visit. During the uptake phase of 45-80 minutes (mean 58.58 ± 8.23), patients were instructed to sit in a quiet room and were encouraged to hydrate and void before the scan. The 3D mode time-of-flight (TOF) Biograph mCT flow (Siemens) was used for PET/CT scanning with low-dose CT for attenuation correction. Patients were positioned supine with both arms raised. A low-dose CT scan (120 kVp, 35 mA, slice 0.6 mm) from the base of the skull to the mid- thighs was obtained before the PET scan. The CT image data were then automatically used to position the patient for the PET acquisition (scan duration 12 s, image size 400, FWHM 3mm). For processing, TrueX+TOF (ultraHD-PET) reconstruction method was applied, using 21 subsets and 2 iterations.

Using a dedicated Syingo Via workstation, all PET images were interpreted by two nuclear medicine physicians experienced in reporting PET/CT imaging studies. Both physicians blindly and independently evaluated the images to eliminate any bias in the results. For quantitative analysis of a background region in the body, maximum standardized uptake value (SUVmax) of the liver FDG uptake was calculated and recorded. For qualitative analysis, a visual scoring of image quality on a scale of 1 - 3 (good,1; acceptable,2; poor,3) was performed on all images and results were compared between images of weight and BMI 18F-FDG adjusted dose. SPSS version 23 software was used as a statistical tool. P-value 0.05 or less has taken as significant. Kappa test was used to evaluate the agreement between the two observers.

3. RESULTS

Based on patients BMI, 6.5% (2 patients) were categorized as underweight, 48.4% (15 patients) as normal weight, 22.6% (7 patients) as over-weight, and 22.6% (7 patients) as obese. Patients had mean glucose level of 5.37± 0.767. Table 1. presents the prescribed 18F-FDG doses to patients based on their weight and based on their BMI in the follow up scan.

TABLE 1. 18F-FDG INJECTED DOSES BASED ON WEIGHT AND BMI CALCULATIONS

Injected dose 18F-FDG	Weight- adjusted	BMI- adjusted
Min	6.10 mCi	4.51 mCi
Max	12.11 mCi	9.65 mCi
Mean	9.1 mCi	6.55 mCi
SD	1.41	1.14

An average of 26.9% ± 15.3% dose reduction with a maximum of 49.9% (5.57 mCi) dose reduction was observed when injected dose were based on the BMI of patients. Out of the total dose reduction, 49.1% dose reduction was observed with underweight patients, 31.2% with normal weight patients, 19.8% with overweight patients, and 18.5% with obese patients as seen in Fig 1.

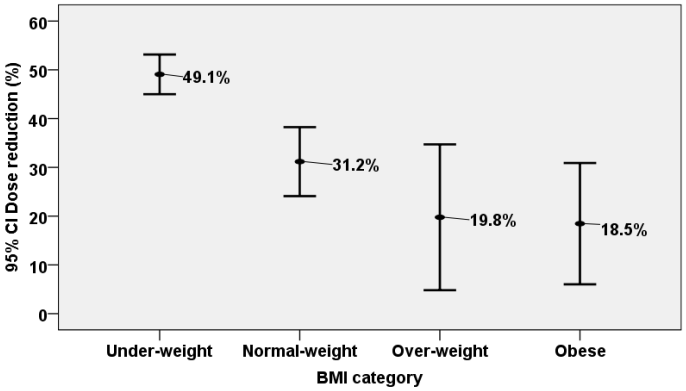

FIG. 1. Percentage of dose reduction in relation to BMI category

According to the two observers scoring of the 62 images (31 images based on weight adjusted dose and 31 images based on BMI adjusted dose), good images were only observed with normal weight patients, where as poor images were observed with normal weight, over weight and obese patients. A stronger level of scoring agreement was observed between the two evaluators with images based on BMI adjusted dose [kappa value = 0.603 (P < 0.0005)] compared to images based on weight adjusted dose [kappa value = 0.473 (P= 0.001)]. By comparing SUVmax of both sets of images, SUVmax (mean 3.36 by observer A, 3.15 by observer B, mean 3.25 +/-0.65 by both observers) of weight-adjusted dose images was comparable to SUVmax (mean 3.19 with observer A, 3.12 with observer B, mean 3.15+/-0.747 with both observers) of BMI-adjusted dose images as seen in Fig 2.

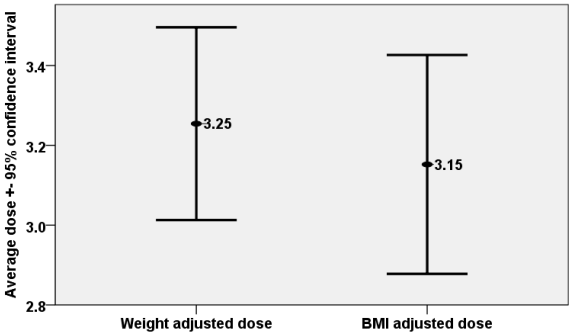

FIG. 2. Mean SUVmax of weigh and BMI adjusted dose images

4. DISCUSSION

According to the findings of this study, injected dose of 18F-FDG to patients was lowered after introducing the BMI adjusted dose protocol. Visual inspection of PET images is the main tool for image interpretation. Despite that more than one quarter of the weight adjust dose was reduced when injected by BMI adjusted dose, there was no significant difference in the quality of both sets of images.

Majority of image scoring was acceptable in all four BMI categories. Good scoring of the images was observed with only normal weight patients where as poor scoring was observed with normal weigh, overweight and obese

patients. Maintained image quality was as well influenced by the use of 3D acquisition mode [1] and the technology of TOF in the PET/CT scanner. With this technology, more data is collected by the PET detector from the photons that travel in the patient's body. This improves image quality while allowing reducing administered dose activity to the patient. In return, it indirectly reduces the patient effective dose [8,9].

The quantitative evaluation of PET images using the standardized uptake value (SUVmax) has been widely recognized as an important tool for the diagnosis, staging and monitoring in oncology [3,10]. Several studies ruled out the effect of injected activity in relation to SUV values [11,12,13]. Regarding image quantification in this study, despite the reduction in BMI adjusted injected dose, the SUV-based analysis of the liver showed no significant difference between images of both methods of 18F-FDG dose calculations. This might as well been influenced by keeping all acquisition and technical parameters of patient preparation constant to exclude any other factors that might affect the outcome of SUV [7].

In overall, qualitative and quantitative analysis of this study proved moving from weight adjusted dose to BMI adjusted dose is to the advantage of the patient. Thus, maintaining good image quality while reducing injected dose fulfills the optimization principle of radiation protection as recommended by the ICRP 60.

5. CONCLUSION

Exploring protocols to optimize patient radiation exposure is important in any radiation protection program. Initial results in this pilot study demonstrated an average of 26.9% reduction in patients administered dose when moving from weight adjusted 18F-FDG dose to body mass index (BMI) adjusted dose. This will potentially minimize effective doses to patients while maintaining good image quality. Moreover, this will reduce radiation exposure to PET paramedics as well as will lead to economic savings. In future, more work needs to be done on patient and staff dosimetry and to validate the proposed protocol with a larger population.

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RISK ANALYSIS IN THERAPEUTIC NUCLEAR MEDICINE

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Abstract

There are few incident reports in therapeutic nuclear medicine and the lessons learned from several radiological events show a fragile safety culture in the organizations. Taking into account the Bonn Call for Action, it is applied the radiological risk analysis for this field, using the half-quantity methodology of risk matrix. The code SEVRRA developed for the Forum of Ibero-American Regulators for diagnostic nuclear medicine is adapted to convectional therapy and specific patient therapy in independent way, with previous study of each one for the technique of failure modes and effects analysis (FMEA). For analysis and treatment risks is used the Cuban code SECURE-MR. In each case, initiators events for specific treatment, direct barriers, frequency reducers and consequences reducers are identified, for a hypothetic nuclear medicine service with two kinds of therapies. The paper shows that a 6.90% and 9.94% of high risk are obtained for each one practice. These are reduced to 0.69% and 0.58%, respectively. The steps of process with the major contribution to the inacceptable risk are identified as the radiopharmaceuticals preparation and the maintaining of equipments and systems. For conventional therapy there is mainly medium impact for workers and public, in the first case with higher incidence. In 12.3% and 1.8% of studied occurrences, there are very high and high consequences, respectively, for patients. Nevertheless, in the specific treatment, there is a similar behaviour of the high consequences but this is increased to the higher level of 22.3%. Barriers, frequency reducers and consequences reducers with higher impact to risk are identified in spite of their specific participation (less than 11%). The importance of incidental factors in adequate performance of the human behaviour in the radiological occurrences is established and this is an experience that could be applied for the improvement of radiological safety in these practices.

1. INTRODUCTION

The establishment of a quality management system in any activity or process requires today the risk management [1], and for this purposes the referents [2-3] are methodological guides. In the Second Conference on Radiation Protection in Medicine (2012), there is a calling action form the International Atomic Energy Agency (IAEA) and The World Health Organization of (WHO) [4] which establishes that should be implemented prospective risk analysis methods to enhance safety in clinical practice.

There is no reference on this kind of systemic study in therapeutic nuclear medicine including all of the possible occurrences from the design to radioactive wastes management taking into account the following surveillance of patients. Many radiological events are not reported and indeed the necessity to study for determining the influence of human and equipment failures. For instance, it will be possible to know how much the dose calibrator failure allows a higher risk for patient. With this tool, there is a way to determine the main actions for reducing the occurrence of this kind of events. There may be additional latent risks for accidental exposures, which have not been reported or have not occurred, but are possible and may occur in the future if not identified, analyzed, and prevented by safety provisions.

2. MATERIALS ANDMETHODS

A list of ‘‘initiating events’’ is conformed from study reports [5-8]. For convectional therapeutic nuclear medicine (CTNM) and specific patient therapeutic nuclear medicine (EPTNM) a generic model is conformed in both cases. The FMEA analysis technique for each step’s process is used with adapted numerical scale reported

in [9] for workers and public. The risk matrix approach is applied considering all basic aspects [9]. As with any anticipative method, the risk matrix involves a systematic search for potential risks; that is, any situation that can cause an accidental exposure. The method contributes new insights: The application of the risk matrix approach has identified that another group of less catastrophic but still severe single-patient events may have a higher probability, resulting in higher risk. In addition, the possible events with consequences to workers and public are identified. Both models are inserted in the software SEVRRA for future analyses in hospitals. Risk analysis is carried out with the Cuban code SECURE-MR which allowing treatment of high level risk. The defence in depth principle is applied. There is defined robustness for each control risk element as soft, normal and robust.

3. RESULTS ANDDISCUSSION

For CTNM 145 total ‘‘initiating events’’ (IE), 121 barriers (B), 67 frequency reducers (FR) and 30 consequences reducers (CR) are identified. Furthermore, for EPTNM there are 171 IE, 152 B, 73 FR and 33 CR. Comparative pictures with obtained results without the influence of control elements for CTNM and EPTNM are show in FIG.1. The second was screening taking into account all barriers, frequency reducers and consequences reducers and their incidences. This can see in FIG. 2.

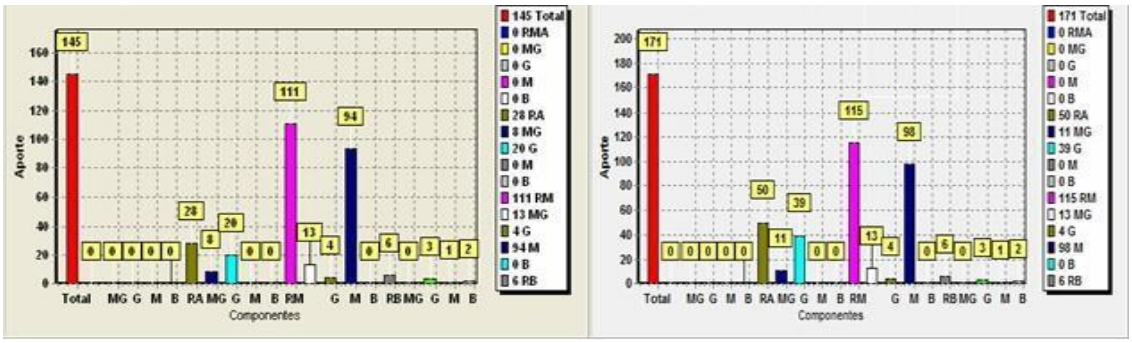

FIG. 1. Risk profile in CTNM (left) and EPTNM (right) without the influence of control elements.

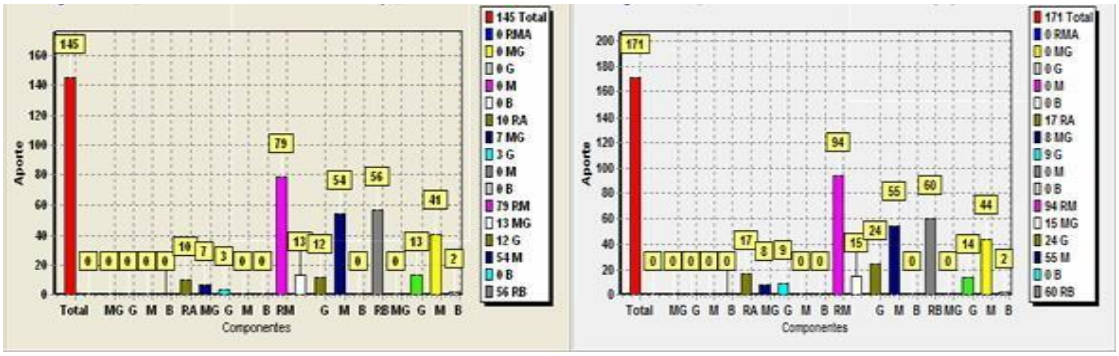

FIG. 2. Risk profile in CTNM (left) and EPTNM (right) with the influence of control elements and their robustness.

TABLE 1 and TABLE 1 show risk levels by each step in the two studied practices and FIG.3 visualizes the behaviour of risk level when adopted measures are introduced. As it can see, the high risk (HR) is reduced to 0.58%. Only is resting the event in the oncological surveillance of patients.

When there is a failure of the dose calibrator the high risk is increased to 10% of this for CTNM and 11.76% for EPTNM. The FIG. 4 reflects medium impacts for workers and public, in the first case with higher incidence. In CTNM there are very high consequences (VHC) for patients in 12.3% of accidental sequences. These represent a 1.8% of high consequences for patients. For EPTNM patients have 11.9 % of VHC) and 22.3% of high consequences (HC), respectively. This is a foreseeable behaviour.

TABLE 1. RISK PROFILE BY EACH STEP IN CTNM.

Process step	VHR	HR	MR	L	Total
Design of the nuclear medicine service	0	0	10	18	28
Construction of the nuclear medicine service	0	0	5	10	15
Acceptance and setting in service of the used equipments	0	0	5	3	8
Maintenance and fixing of equipments and systems	0	3	4	1	8
Clinical patient assessment	0	0	1	0	1
Clinical treatment prescription	0	2	4	6	12
Reception of radiopharmaceutical	0	0	9	2	11
Preparation of radiopharmaceutical	0	4	4	3	11
Transferred dose to the administration	0	0	3	0	3
Administration of radiopharmaceutical	0	0	6	9	15
Patient internment	0	0	10	3	13
Ambulatory treatment	0	0	1	0	1
File and surrenders of the results of the treatments	0	0	1	0	1
Radioactive waste management	0	0	15	1	16
Oncological surveillance of patients	0	1	1	0	2
Process	0	**10**	79	56	145

TABLE 2. RISK BY EACH STEEP IN EPTNM.

Process step	VHR	HR	MR	L	Total
Design of the nuclear medicine service	0	0	10	18	28
Construction of the nuclear medicine service	0	0	5	10	15
Acceptance and setting in service of the used equipments	0	0	5	3	8
Maintenance and fixing of equipments and systems	0	3	3	2	8
Elaboration of the treatment protocol for clinical assay	0	0	1	2	3
Clinical patient assessment	0	0	1	0	1
Planning of the treatment	0	2	7	0	9
Clinical treatment prescription	0	1	5	6	12
Reception of radiopharmaceutical	0	0	9	2	11
Preparation of radiopharmaceutical	0	4	4	3	11
Transferred dose to the administration	0	0	3	0	3
Administration of radiopharmaceutical	0	0	6	9	15
Patient internment	0	0	10	3	13
Ambulatory treatment	0	0	1	0	1
Post-treatment acquisition of images	0	2	6	1	9
Processing of treatment images	0	2	0	0	2
Interpretation of the images and formulation of results	0	2	1	0	3
File and surrenders of the results of the treatments	0	0	1	0	1
Radioactive waste management	0	0	15	1	16
Oncological surveillance of patients	0	1	1	0	2
Process	0	**17**	94	60	171

The importance of barrier, frequency reducers and consequences reducers are reflected in FIG. 5-7. Each code identified each measure and there is knowledge what doing to prioritize for the patient, workers and public radiological safety.

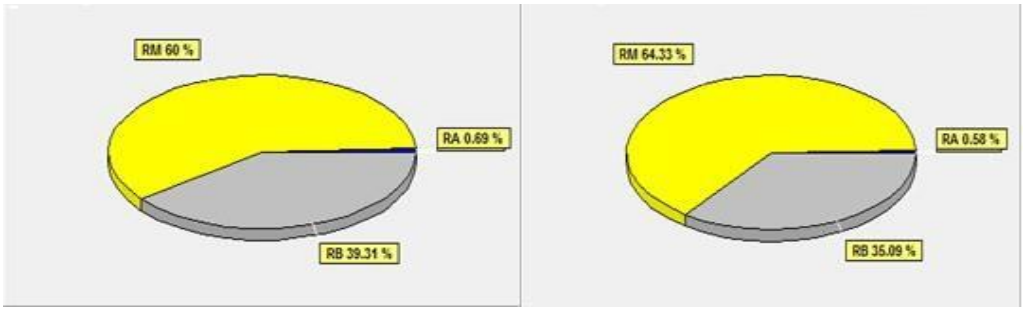

FIG. 3. Risk profile in CTNM (left) and EPTNM (right) with the treatment of high level of risk

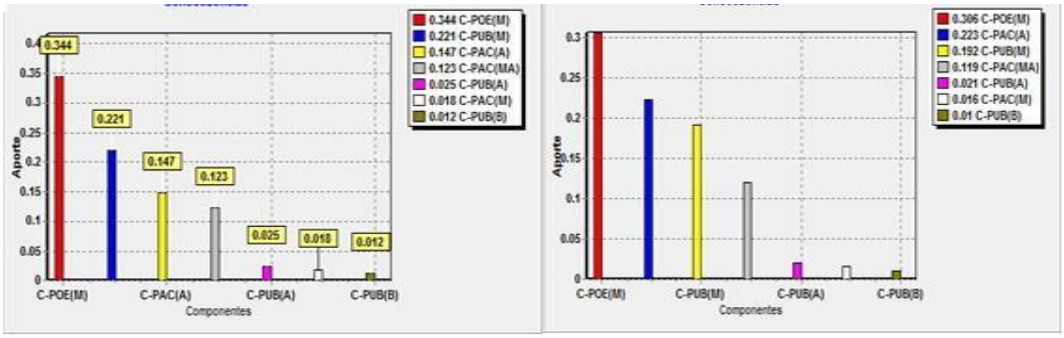

FIG. 4. Level of consequences for patients, workers and public in CTNM (left) and EPTNM (right).

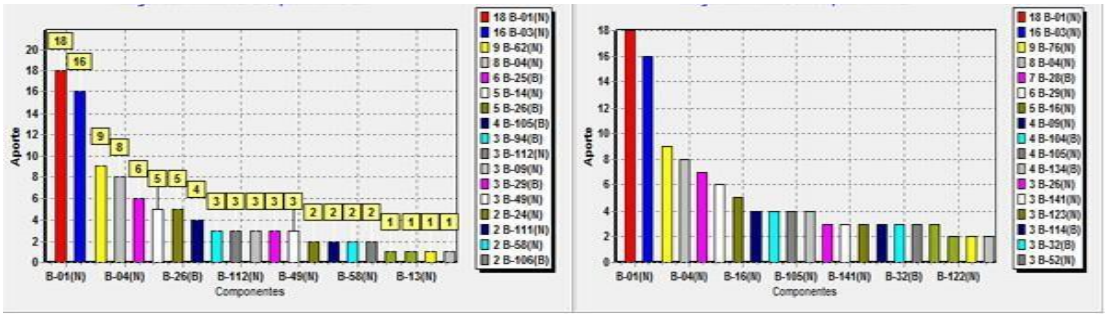

FIG.5. Barrier influence in the accidental sequences in CTNM (left) and EPTNM (right).

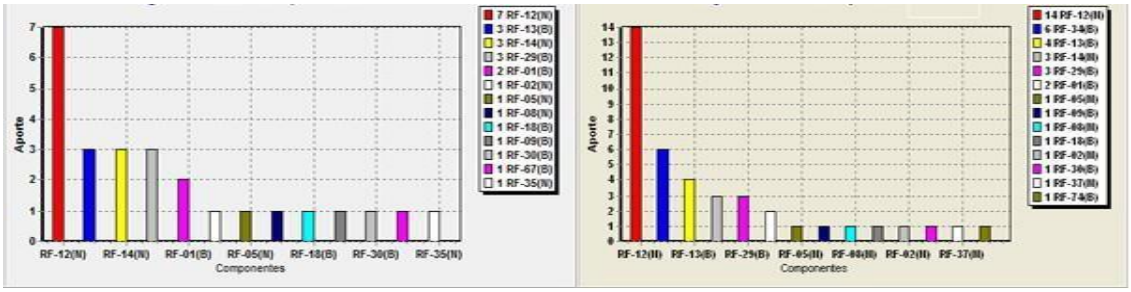

FIG.6. Frequency reducer influence in the accidental sequences in CTNM (left) and EPTNM (right).

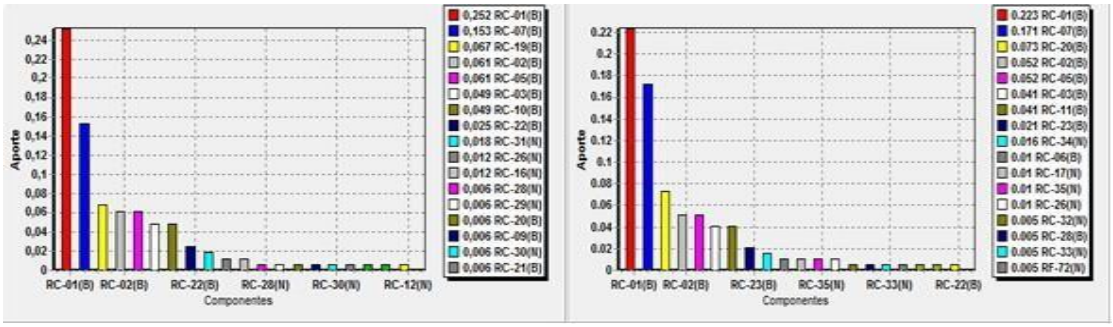

FIG.7. Consequences influence in the accidental sequences in CTNM (left) and EPTNM (right).

4. CONCLUSIONS

The analysis of all possible events in the CTNM and EPTNM show for the two generic simulated models there is not a very high level of risk, but should be taken reducing high-level actions. These represent 6.90% and 9.94%, respectively of the total postulated occurrences. Equipment failure is 12 % of the high risk while human error belongs to 88%. Barriers which have the most incidence in the risk level are applied regulatory documents revision for design and construction, initial radiological assessment of different areas, measurement of radiopharmaceutical activity with double checks to avoid or detect an error (for another person and dose calibrator) and patient document revision for Medical Committee before the treatment. The major impact preventive measures are identified as a moderate workload, education and training activities for the specialist who executes the project revisions and the radiation protection officer. For the ETNM they are included for the medical physic. For consequences reducers the periodical monitoring of areas, emergency procedures for decreasing of critical organs dose due to mistaken radiopharmaceutical administration and periodical revision of treated patient cases are the most important for risk. All of these measures allowance an adequate control risk of these practices. This experience may be useful for other places in this field towards the radiological safety improvement and a safety culture development with a report culture and the lesson learned. The used tool and generic models provides an opportunity for self-evaluation and managing the safety measures that are most suitable to the hospital's own conditions.

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SECOND NATIONAL SURVEY OF PATIENT DOSES FROM PET/CT EXAMINATIONS IN BULGARIA

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Abstract

The first national survey of patient doses from PET/CT was performed in Bulgaria in 2013-2014. It included both systems available in the country. New equipment was installed in recent years prompting the Bulgarian Society of Nuclear Medicine to initiate a second national survey. The aim of the paper is to estimate patient exposures from all six PET/CT systems and to suggest optimization measures. Data for a total of 453 patients undergoing either whole body or epilepsy (on one system) examination with ^{18}F-FDG were reported. Effective doses from the CT component were calculated using commercial software. The radiopharmaceutical (RPh) contribution was estimated by applying ICRP Publication 128 conversion coefficients. CT effective doses from whole body examinations on the six systems were as follows: 8.1, 7.1, 3.6, 9.0, 2.0 and 2.7 mSv. The RPh contribution was 3.9, 4.6, 4.5, 4.1, 4.4 and 4.2 mSv respectively. The mean activities applied on both systems participating in the first survey and hence effective dose from RPh were decreased by 20% and 22% respectively. Patients examined with the new equipment received about one and half to two times less exposure compared to the old systems. Further measures for optimization were considered.

1. INTRODUCTION

The first two positron emission tomography/computed tomography (PET/CT) systems were installed in Bulgaria in 2009-2011, followed by the first survey of patient doses from hybrid imaging in 2013-2014 [1]. Afterwards new equipment was installed prompting the Bulgarian Society of Nuclear Medicine to initiate a second national survey with the aim to estimate patient exposures from all six PET/CT systems and to suggest optimization measures.

2. MATERIALS AND METHODS

The systems included in the survey were: Gemini TF (Philips Healthcare), denoted as SI, Discovery 600 (GE Healthcare), denoted as SII, both participating in the first survey also, Discovery IQ Clarity 5 ring (GE Healthcare), denoted as SIII, Biograph Duo (Siemens Medical Solutions), denoted as SIV, and two Biograph mCT64 (Siemens Healthineers), denoted as SV and SVI. SI, SII and SIII are equipped with 16-detector row CT, SV and SVI include 64-detector row CT, while SIV is equipped with 2-detector row CT. Patient data were retrospectively collected including age, sex, height, weight, CT exposure data (tube voltage, rotation time, collimated beam width, pitch, use of tube current modulation (TMC), computed tomography dose index (CTDI), dose length product (DLP)), activity applied, and scanned region of the body. All examinations were performed with the radiopharmaceutical (RPh) ^{18}F-2-fluoro-2-deoxy-D-glucose (FDG).

The same methodology as the used in the first national survey with one small correction was applied for consistency. This small difference will be discussed in the next section. Patient data were separated by sex and averaged CTDI, DLP and RPh activities were used for estimation of effective dose. The CT contribution was calculated with CT Expo software (Medizinische Hochschule, Hannover, Germany), introducing averaged CTDI and DLP, also the CT scanner model, exposure data and scanned region of the body. The separately calculated by gender effective dose was averaged over both sexes afterwards. Calculations were performed based on both ICRP Publication 60 [2] and ICRP Publication 103 [3] tissue weighting factors, provided by the software. The RPh contribution was estimated by applying ICRP Publication 128 conversion coefficient on the averaged activity [4]. The conversion coefficient for ^{18}F-FDG has the same value as the published one previously [5]. New conversion coefficient provided by Andersson et al. was also used for comparison [6].

3. RESULTS AND DISCUSSION

The second survey included data for a total of 453 patients undergoing either standard whole body or epilepsy (only on SII) examination with ^{18}F-FDG. The patient statistics including the number of patients, administered activity, CTDI and DLP are presented in Table 1. The data from the first survey are also included, denoted as “old”, while data from the current survey are denoted as “new” [1]. Mean values, minimum, maximum and standard deviation (SD) of the quantities are shown.

TABLE 1. NUMBERS OF PATIENTS AND STATISTICAL DATA ON ADMINISTERED ACTIVITY, CTDI AND DLP PER TYPE OF EXAMINATION FOR THE OLD [1] AND THE NEW SURVEYS

Examination	PET-CT system	No patients	A_{mean} (min, max); SD (MBq)	$CTDI_{mean}$ (min, max); SD (mGy)	DLP_{mean} (min, max); SD (mGy.cm)
Standard	I/old	43	258 (185, 418); 55.8	5.9 (4.2, 8.5); 1	591 (395, 1012); 130
Standard	I/new	45	205 (159, 270); 29	5.8 (3.5, 7); 1	558 (329, 773); 113
Standard	II/old	52	313 (153, 511); 73	5.4 (1.8, 11.1); 2	575 (194, 1242); 248
Standard	II/new	50	243 (171, 345); 45	5.1 (2.2, 8.9); 1.9	545 (225, 923); 211
Epilepsy	II/new	17	202 (91, 530); 85	2.3 (1.8, 4.2); 0.7	66 (54, 114); 18
Standard	III	99	238 (168, 333); 41	2.6 (1.3, 6.1); 1.1	279 (137, 651); 120
Standard	IV	51	216 (128, 299); 42	6.6 (5.5, 9.7); 1.2	625 (494, 994); 125
Standard	V	40	230 (175, 350); 40	1.5 (1, 3.6); 0.5	138 (83, 306); 44
Standard	VI	56	222 (175, 286); 28	2 (1.4, 3.4); 0.4	209 (134, 342); 45

The main technical parameters for CT scanning are provided in Table 2. TCM depicts whether tube current modulation is used or not. The mean scan range in mm, introduced in CT Expo per gender, is also included. The latter is chosen depending on the description of the scanned body region, provided by the local staff. For the standard examination there were small differences, describing scanning either from top of the head or from the end of the nose, till mid thighs or till just below the trunk.

TABLE 2. MAIN EXPOSURE DATA FOR CT SCANNING

Examination	PET-CT system	kV	Rotation time (s)	Collimated beam width (No. rows x mm)	Pitch	TCM	Mean scan range in software (mm) Males/Females
Standard	I/old	120	0.5	24	0.69	Manual	113/104
Standard	I/new	120	0.5	24	0.813	Manual	101/93
Standard	II/old	120	0.8	3.75	not provided	TCM	113/104
Standard	II/new	120	0.8	20	1.375	TCM	113/104
Epilepsy	II/new	120	0.8	10	1.375	TCM	24/24
Standard	III	120	0.6	20	1.375	TCM	113/104
Standard	IV	130	0.8	not provided	not provided	Manual	113/104
Standard	V	120	0.5	19.2	1.2	TCM	98/90
Standard	VI	120	0.5	19.2	1.4	TCM	113/104

The calculated effective doses in mSv are presented in Fig. 1. The CT contribution based on ICRP 60 [2] calculations is presented in the first column for all systems and surveys, the second column is CT contribution based on ICRP 103 [3] calculation, the third column is the RPh contribution based on ICRP 128 [4] calculation, the fourth column shows RPh contribution based on Andersson et al. coefficient [6], and the fifth column is the total effective dose, based on ICRP 103 CT calculation and ICRP 128 RPh contribution. Data from the first survey are semi-transparent for better visualization. Numerical values are shown for the calculations, used for total effective dose estimation.

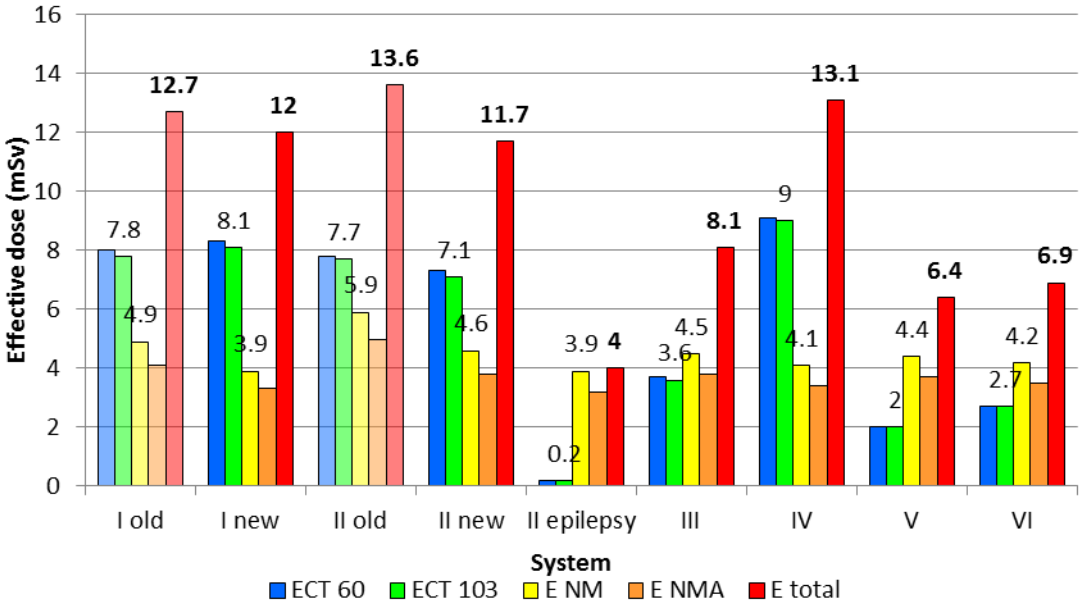

FIG. 1. Effective dose (E) estimations. The first blue and the second green columns represent E from the CT, applying ICRP 60 [2] or ICRP 103 [3] tissue weighting factors respectively, the third yellow and the forth orange columns show E from the RPh based on ICRP 128 [4] or on Andersson et al. [6] conversion coefficients respectively, and the fifth red column shows E from the whole examination, taking into account CT calculation based on ICRP 103 and RPh calculation based on ICRP 128.

A difference in the methodology used in the present survey compared to the previous one is the choice of patient weight. In this study patients with weights between 50 and 100 kg were selected. In the first survey all patient data were included. Careful data examination showed that only few patients were included with weights

above 100 kg or below 50 kg almost compensating the weight difference effect in both surveys. Decrease of the mean administered activities and hence of effective dose from the RPh of 20 % for SI and of 22 % for SII was found (Table 1 and Fig. 1). Slight decrease of CT exposure on SII was observed that could be attributed to difference in patient cohorts. Total effective dose from epilepsy examination was about three times lower (4 mSv) in comparison to the whole body examination (11.7 mSv) on the same PET/CT scanner.

Although purchased after the first national survey, SIV is a second hand mobile PET/CT, manufactured in 2006. Doses observed on this system (9 mSv from CT and 4.1 mSv from RPh, total effective dose 13.1 mSv) were comparable to doses from the second survey on SI (8.1 mSv from CT, 3.9 mSv from RPh, total 12 mSv) and SII (7.1 mSv from CT, 4.6 mSv from RPh, total 11.7 mSv), that are also relatively old scanners, installed before 2011 (Fig. 1). The last generation systems SIII, SV and SVI exhibited the opportunity of the new equipment to use low dose levels to achieve the diagnostic task (the total effective dose on SIII is 8.1 mSv, on SV it is 6.4 mSv and on SVI it is 6.9 mSv). On SIII new iterative reconstruction software on the CT is planned to be installed in near future. Further decrease of CT doses is expected thereafter. In general patients examined with the new equipment received about one and half to two times less exposure than those examined on the older scanners. However other possibilities for optimization could be searched. SV and SVI are the same type, but the latter delivers about 26 % higher dose from the CT (2 mSv on SV and 2.7 mSv on SVI, Fig. 1). A survey performed in France in 2011 reported the mean effective dose from whole body PET/CT to be 14.3 mSv, which is higher than all estimations in the present study[7].

Effective doses from the RPh calculated based on Andersson et al. conversion coefficient were about 16 % lower compared to ICRP 128 estimation [4, 6]. It is expected that the former method is more precise, taking into account that the coefficients are calculated based on more realistic voxel phantoms, new decay data, and ICRP Publication 103 tissue weighting factors [2]. Consecutively the effective doses from PET/CT imaging based on current ICRP coefficients areoverestimated.

Limitation of this study is the lack of comparison of image quality. Future surveys should ideally include such criteria.

4. CONCLUSIONS

Second national survey of patient exposures from PET/CT imaging was conducted. Some decrease of exposure from the RPh of about 20 % compared to the first survey was observed. The new equipment installed in Bulgaria exhibited significant potential for dose saving. Additional optimization could be achieved. Total mean effective dose from whole body PET/CT varied between 6.4 and 13.1 mSv and for epilepsy it was 4 mSv. More realistic dose estimations are expected based on the new conversion coefficients, provided by Andersson et al. [6].

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Residual Dose Rates After Radiactive Iodine -131 Treatment In The Day Of Hospital Discharge

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Abstract:

Turkish Atomic Energy Authority has its own regulations for the patients who were undergone radioactive treatments. Our aim is to investigate the discharge dose rates of patients who were administered 3700 MBq I-131 treatment for thyroid cancer.

Patients of Methods:

A total of 91 patients (64 female, 27 male) who were totally thyroidectomised with a diagnosis of well differentiated thyroid cancer , previously and administered 3700 MBq I-131 , hospitalized for two days between 2011 and 2013 included to the study. Their dose rates on the time of discharge were evaluated, statistically

Results :

Discharge dose rate mean value 1.75± 0.89 mR/h (17,5μSv/h±8,9) and the mean restriction period for public radioprotection were 13,44±4.02 days. On the day of discharge 3 patients dose rates were above the limits. 8 patients dose rate were equal to legislation limits. Those patients who had high dose rates were not discharge until their those rate decreased. The mean restriction period for public relations was 13,44 ± 4.02 days . Mean residual activity measured at the discharge 307,1±156,14 MBq. In our group of patients %88 of patients had lower dose rates than the limits after fourty eight hours hospitalization.

Discussion:

Medical radiation exposures are applied to patients when benefits of treatment overcome the damages. Radiation dose that is received by people during exposure after treatment with radionuclides, countermeasures must be taken.

INTRODUCTION

Thyroid cancer is the most common endocrine cancer (approximately 1.0%–1.5% of all new cancers diagnosed each year in the USA) [1], and its incidence has continuously increased in the last three decades all over the world.

Radioactive iodine treatment for people suffering welll differentiated thyroid cancer after bilaterally total thyroidectomy is a well known , broadly used treatment. For almost five decades, radioiodine treatment of thyroid cancer has been based empirically on administered activity for thyroid remnants in patients with thyroid cancer. Treatment usually comprises thyroidectomy, and, when indicated, adjuvant radioactive iodine (RAI) therapy. Decision-making regarding the extent of surgery remains a subject of debate, reflecting a lack of high-quality data to support management decisions. Similar levels of controversy exist regarding the role of adjuvant RAI therapy. The administration of adjuvant RAI has 2 main objectives: first, to ablate occult microscopic foci of WDTC, reducing the risk of tumor recurrence, and, second, to ablate any remaining normal thyroid tissue, thereby facilitating surveillance with serum thyroglobulin or radioiodine whole-body scintigraphy.(2) Radioactive iodine treatments main objective is destroying possible thyroid remnants after surgery, in addition to any other possible thyroid cancer cells that have metastasized. On the basis of these objectives, the decision to use RAI depends on the risk of the original thyroid cancer and the completeness of surgery in removing malignant and normal thyroid tissue.(3)

Our aim is to investigate the discharge dose rates of patients who were administered 3700 MBq I-131 treatment for thyroid cancer, in this retrospective study.

PATIENTS AND METHODS:

A total of 91 patients (64 female, 27 male) who were previously total thyroidectomised with a history of well differentiated thyroid cancer and administered 3700 MBq I-131 , hospitalized for two nights between 2011 and 2013 included to the study. All patients were advised exess water consumption, having shower, frequent urination. Their dose rates on the time of discharge were measured by a Geiger Muller detector (Ludlum measurements 14C model, 2000 Texas). Hospital records analysed statistically. Exclusion criteria was renal failure.

RESULTS :

Discharge dose rate mean value was 1.75± 0.89 mR/h (17,5μSv/h±8,9) and the mean restriction period for recommendations up to Turkish Regulations about travel times, close contact, for public radioprotection were 13,44±4.02 days (Table 1). Dose rate of eight patients were equal to legislation limits (3 mR/h). On the day of discharge three patients have dose rates above the limits. Mean residual activity measured at one meter distance at the discharge calculated by the formula ($A.I_{\gamma}/d^2$) measured and was 307,1±156,14 MBq. Their calculated dose rates were 4 (40μSv/h), 4(40μSv/h), and 5 (50μSv/h) mR/h from one

meter. Those patients who had high dose rates were not have been discharged until their those rate decreased.

Patients who had retained dose rates equal to limits, encouraged for a shower, changing clothes and urination. After eight hours , dose rates were measured again at 1 meter distance and when acceptable limits reached <3 mR/h, they were all discharged. The other 3 patients who had high dose rates were also encouraged to drink water more than 2 liters, having a shower , change clothes and frequent urination. The 2 patient whose retained dose value were 4 mR/h had a stay for more 8 hours and the third patient with 5 mR/h stayed 1 more night.

In our group of patients (80/91) of patients had lower dose rates (Table 2) than the limits after forty eight hours hospitalization and . Only 1/91 patients had one more night hospitalization, where two patients had an eight hours of discharge.

DISCUSSION

Recently, an increased incidence for thyroid cancers of all stages (localized, regional, and distant staged disease) has been confirmed . (4) The increased incidence of thyroid cancers of large volume and advanced stages, usually clinically apparent, although the reasons for the worldwide increase in thyroid cancer remains unclear (4,5). However this means increasing need for radioactive iodine -131 treatment up to guidelines. (6)

Among radioactive treatments, high dose radioactive iodine -131 treatment is the most crowded group which is used worldwide. Measures for radioactive treatment have first been implemented with I-131 therapy.

Radioprotection aims to control radiation risks for society without limiting the potential benefits for individuals. Up to Turkish regulations patients discharge rule is achieving 3 mR/h dose rate and commonly an overnight hospitalisation is enough. But in our institution we prefer to hospitalise them for forty eight hours in separate rooms. This may be an advantage for achieving limits. On the other hand this may be argued that long hospitalization period by means of quarantine is not cost effective, the radiation exposure will be high as a consequence of crowded household and young population. So different countermeasures must be taken for different households and communities to avoid risks.

Table 1. Release of patients after radioiodine treatment up to Turkish regulations.

Effective dose rate at 1 meter(μSv/h)	Retained activity	Period for recommendations

<40	<800 MBq	3 weeks
<30	<600 MBq	2.5 weeks
<20	<400 MBq	2 weeks
<10	<200 MBq	1 week
<5	<100 MBq	3 days
<3	<60 MBq	24 hours

Table 2. Retained dose rates after radioactive iodine treatment in our group of patients.

Number of patients	Dose rates after 48 h hospitalization (mR/h)	Delay of discharge
80	1,51 ±0,74	No delay
8	3	No delay
3*	4 4 5	8 hours 8 hours 24 hours

*: Dose rates for each patient

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IODINE-131 THERAPY: 3-YEAR EXPERIENCE IN RADIATION PROTECTION

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Abstract

Iodine-131 is the most commonly used unsealed radionuclide in radiation therapy. Depending on local legislation, patients who receive high activities of I-131 are required to stay in specially designed therapy wards before the residual activity drops below a certain level. This study presents all activities carried out before, during, and after the use of I-131 for therapy purposes. It includes basics of planning the therapy ward, assessment of environmental impact, results of personal dosimetry, evaluation of doses at 1 m distance from the patient, management of radioactive waste, and analysis of some special cases. Prior to introduction of high dose I-131 shielding for the rooms that will accommodate therapy patients was designed, followed by the report where environmental impact of I-131 discharge was evaluated. Daily measurements of ambient dose rate provided valuable data on physiological decay of I-131 for different patient groups. Personal dosimetry of workers was evaluated. Special emphasis was given to the solid radioactive waste generated in the therapy ward. With the appropriate use of radiation protection measures we have managed to keep doses to workers and public at minimum level, and reduce unnecessary risk of radiation exposure.

1. INTRODUCTION

Radiotherapy in nuclear medicine uses radionuclides in a form of specific radiopharmaceuticals to target specific tumours. The most common one is the thyroid carcinoma, but it is used for other purposes too. A pharmaceutical is a carrier of the radioactive isotope with a mission to target specific tumours cells or its metastases. Radiation emitted after the radioactive decay damages tumour cell DNA causing it to die, and consequently the tumour to shrink or disappear. An ideal radiopharmaceutical would be the one that acts exclusively on malignant tumours, anywhere in the human body, while sparing the healthy tissue and organs.

The first radionuclide used for radiation therapy was ^{131}I. It was introduced in 1940s, and since then their range and scope have undergone further expansion [1]. Radionuclides are now used for range of cases, not only for carcinoma [2]. Although some of the applications are now established as a *de facto* standard, the ^{131}I remains the most common radionuclide used in nuclear medicine ablation therapy.

Use of high activity unsealed radiation sources comes with a share of risk. According to the local legislation, patients who receive high activities of ^{131}I are required to stay in specially designed therapy wards before the residual activity drops below a certain level. In March 2014 Clinical Centre of Sarajevo University opened a ward for patients undergoing radionuclide therapy. This paper focuses on radiation protection matters that were considered and all activities carried out before, during, and after the use of ^{131}I for therapy purposes.

2. METHODS

2.1. Planning the therapy ward

In general, the nuclear medicine therapy ward consists of two parts – hot and cold. Hot part is the area with 4 patient rooms (2 beds each), rooms for therapy preparation, application, radioactive waste storage, as well as kitchen and cleaning room. It is separated from the cold part by the locked doors and sanitary passage equipped with hand and foot monitor (Fig. 1). The ward should be able to accommodate patients who received doses up to 7.4 GBq of ^{131}I per session for up to 72 hours after administration. Average dose rate used to calculate the shielding was 0.072 mSv h^{-1}, while dose constraint was 0,3 mSv per year [3].

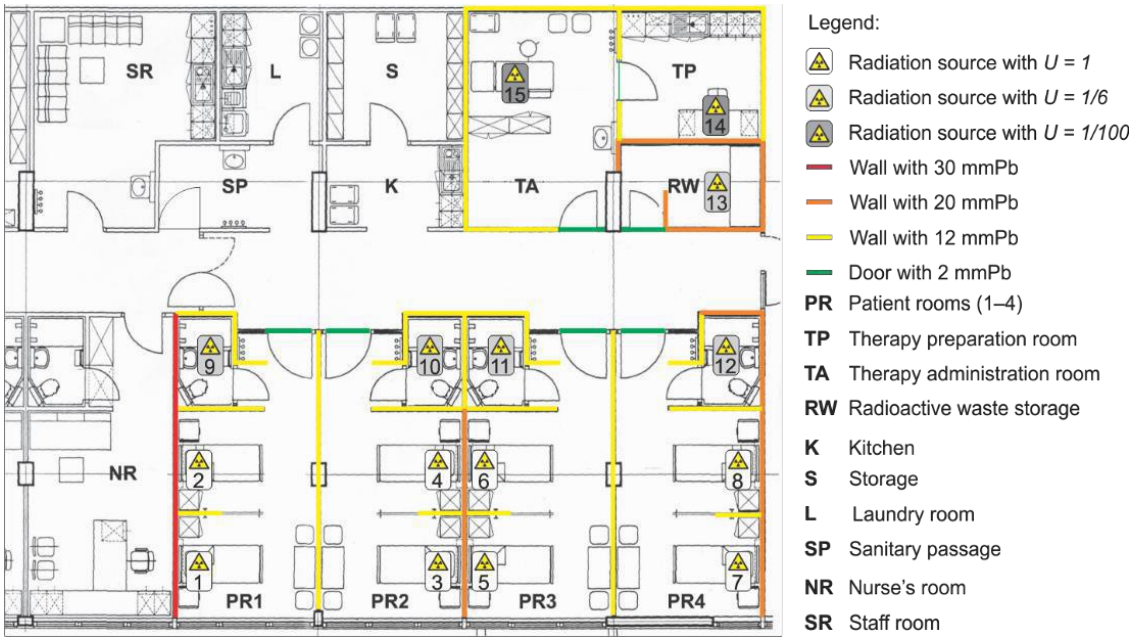

Figure 1. Therapy ward project design. It consists of 4 patient rooms, therapy preparation and administration room, radioactive waste storage, kitchen, general storage, laundry room, sanitary passage, nurse's and staff rom, with different structural shielding (walls and doors) and location of radiation sources with different use factors U.

2.2. Environmental impact assessment

Before any use of unsealed sources in nuclear medicine one needs to calculate impact of radioactive discharge to environment and estimate the effective doses received by population. The methodology used to make the calculations was taken from the International Atomic Energy Agency (IAEA) Safety Reports Series No. 19 "Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances to the Environment" [4].

2.3. Personal dosimetry and dose-rate measurements

Personal dosimetry of exposed workers is performed using thermoluminescent dosimeters (TLD) by an external service. Minimum detectable level (MDL), but also the minimum reportable level set by the local legislation, is 80 μSv per month. The professionals expected to be exposed with higher doses are radiological technologists (assisting the radiopharmaceutical administration) and nurses (24-hour patient care).

Dose-rate measurements at 1 m distance from the patient's chest were performed for each patient treated, every day. For this purpose, we used the Geiger-Müller counter calibrated in ambient dose equivalent rate $\dot{H}^*(10)$ at the energy of 662 keV gamma radiation of ^{137}Cs. Its intrinsic relative error in the measuring range was ±20% max. and energy dependence for gamma radiation dose rate ±30% max. Response time for dose rates above 1 μSv h^{-1} is ≤7 s.

Before measurements patients were asked to take a sitting position on the bed.

2.4. Radioactive waste management

In general, patients generate three types radioactive waste: liquid wastewater, solids contaminated with radioactive isotope and gaseous waste. In accordance to the IAEA position statement, as well as ICRP Publication 94, and after reviewing the safety aspects, we decided to dilute and disperse waste activity, rather than to store it in expensive decay tanks [6]. Solids contaminated by patients (mostly linen and pyjamas) are kept in plastic boxes in the radioactive waste storage room before being washed. This prevents further dissemination of contamination. When it comes to gaseous waste, we considered the current ventilation system to be sufficient.

3. RESULTS AND DISCUSSION

Estimated effective dose to professionals working in the therapy ward with was calculated using a conservative assumption; all beds are occupied 5 days a week with patients who received 7.4 GBq of ^{131}I. Having that in mind, the proposed structural shielding was more than enough to provide adequate radiation protection. Annual effective dose for a nurse working in her/his room 6 hours a day is 0.14 mSv. This adds up to 0.15 mSv she/he would get when working near the patients or in the corridor. All other doses are below dose restriction of 0.3 mSv. As additional safety measure we introduced 10 mm Pb mobile radiation shields which help staff to communicate with patients more easily and without fear of unnecessary exposure.

However, neither were all beds occupied all the time, nor majority of patients received 7.4 GBq. Over the past 3.5 years total number of patients was 303, most of them female (79.8%). Room 1, the one closest to the nurse's room, was occupied with therapy patients only 8 times in 3.5 years. Most of the patients (83.2%) stayed up to 3 days in hospital, leaving the room empty rest of the week. There were only 12 patients who received 7.4 GBq (200 mCi), while the most common dose was 3.7 GBq, prescribed to 57.4% patients (Fig. 2a). As a result, measured personal doses in a year were lower than targeted 0.3 mSv per year. Average dose to therapy ward nurses was 0.1 mSv per year.

Daily measurements of ambient dose rate show how it changes over time (Fig. 2b). $\dot{H}^*(10)$ for patients who received 3.7 GBq on the first day was 0.15±0.03 mSv h^{-1}. The graph clearly shows how disperse the measurements are (Fig. 3). Large standard deviation indicates magnitude of uncertainties involved. We can easily identify few: instrument errors, long response time, distance to patient and its body positioning, different physiology of patients, differences in actual vs. prescribed activity etc.

The average dose rate for all patients over a week (5 days) was 0.030 mSv h^{-1}, while in the first three days the average was 0.048 mSv h^{-1}. This is considerably less than 0.072 mSv h^{-1} taken for the purpose of shielding calculations.

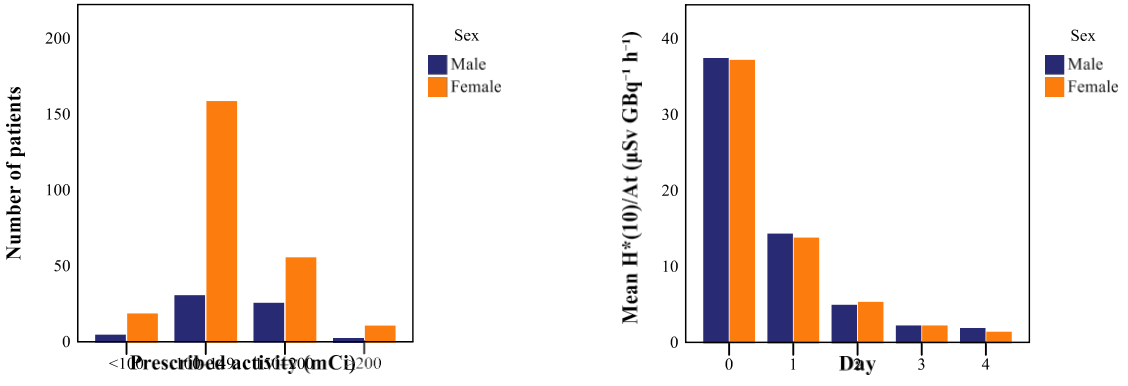

Figure 2. (a) Number of patients with different prescribed activity. (b) Ambient dose rate $\dot{H}^*(10)$ (μSv h^{-1}) at 1 m *distance from patients who were prescribed* 3.7 GBq *of* ^{131}I. *Full line is the exponential regression line* $y = 156.4 \exp(-1.125x)$, *while dashed lines represent upper and lower 95% confidence limits.*

4. CONCLUSIONS

Shielding calculations have been proven to be very conservative. However, top-to-bottom construction of therapy ward in nuclear medicine is something done very seldom. A cautious approach is necessary as one is not aware of possible changes that could happen in future. More accurate results can be used in case when existing rooms are adapted, or when temporary or provisional solutions are required.

Figure 3.

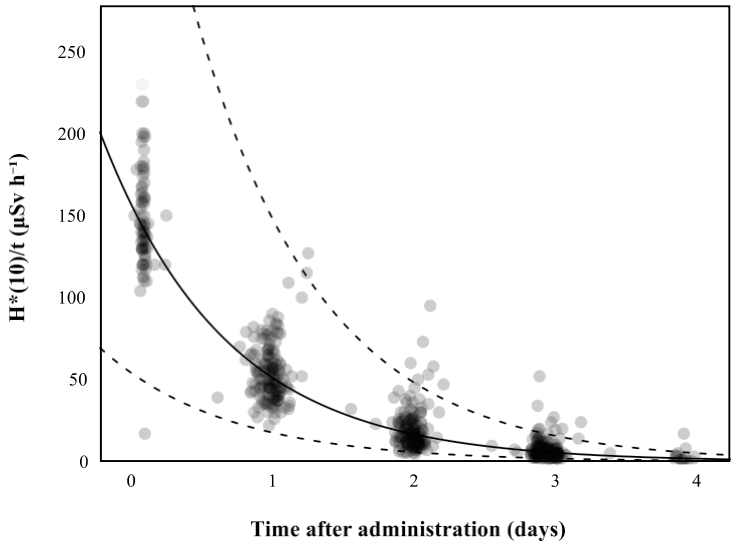

Figure 4. Mean ambient dose rate per unit of activity (μSv GBq^{-1} h^{-1}) *each day after administration depending on patient sex. No significant differences exist* ($p > 0.05$).

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ESTIMATION OF EXISTING STRUCTURAL SHIELDING TO PROVIDE RADIATION PROTECTION FOR IODINE-131 THERAPEUTIC ROOM

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Abstract

The calculation of the correct structural shielding in radiation protection is a premise, and sometimes is not possible to use only the protocols and it is necessary to apply different methods to obtain desired results. In this paper the structural shielding for rooms to treat patients with iodine-131 with 1,110MBq (30mCi) doses or above were calculated. Like in old hospital in developing countries, there are buildings but no the specification of the existents structural shielding. Estimation was carried out in a practical way: using a punctual source and measuring and detecting at different points to get the real structural shielding values of the existing walls. From such attenuation measurements, validated for a company registered in the nuclear security commission of our country, it was able to calculate the necessary barrier to increase the existing ones to be into the required standards. International protocols were used for calculations. The final purpose is to provide a guide for medical physicists in similar situations and to avoid damage to the structure of the building when trying to get its composition. The budget of theproject is reduced as well, since the existing natural structural shielding is used to contain the radiation.

1. INTRODUCTION

It is very frequent in developing countries, for health buildings to be very old that is impossible to find a register and/or architectural plans with details about the composition and not even the thickness of the walls. Such health clinics and hospitals have been constructed without planning for the future investments in expanding their installations. Particularly with the old ones, the main problem is that in the past it was possible to construct without allowance from the regulatory organs, or the laws and the technical regulations were less restricted than nowadays.

Expansion involving new areas related with the use of radioactive material, in such conditions, is a great challenge for medical physicists, since the international protocols on radiation protection must be followed. In this cases, particular procedures, aside the traditional ones must be applied for appropriate shielding calculation found in the national and international protocols of radiation protection. This challenge was presented in our hospital, the case of a Iodine-131 therapeutic area with doses from 1,110MBq (30mCi) up to 7,400MBq (200mCi); or the corresponding patient equivalent dose rate equal or upper 0.05mSv/h (5mrem/h), measured 1m from the patient [1].

In this particular case, it is possible to calculate the structural shielding under the protocols adopted in our country, like NCRP 147 [2] and NCRP 151 [3], taking into account the attenuation provided for the existing wall. Such consideration reduces considerably the budget and even the investment in a new construction, allowing the expansion and modernization of radioactive installations.

This paper has the purpose to facilitate the activity of the medical physicist, who works with radiation protection, in the calculation of structural shielding for radiation protection in the case of existing buildings with no knowledge of actual construction walls.

2. METHODS

Definition of the structural shielding for Iodine-131 (I-131) therapeutic room was conducted at University Hospital of Saltillo (HUS) of Autonomous University of Coahuila, Mexico. The Nuclear Medicine Department of HUS was established in 2006 for diagnosis and treatments under 1,110MBq (30mCi). Nowadays, it is necessary to expand the department and start treating patients with doses from 1,110MBq (30 mCi) up to 7,400MBq (200mCi).

In the context of this kind of treatment, with I-131 at such doses, is mandatory to have a special area for keeping in control those patients. In HUS, the therapeutic room is located in the third floor of the hospital building, in the closed area depicted in Fig. 1.

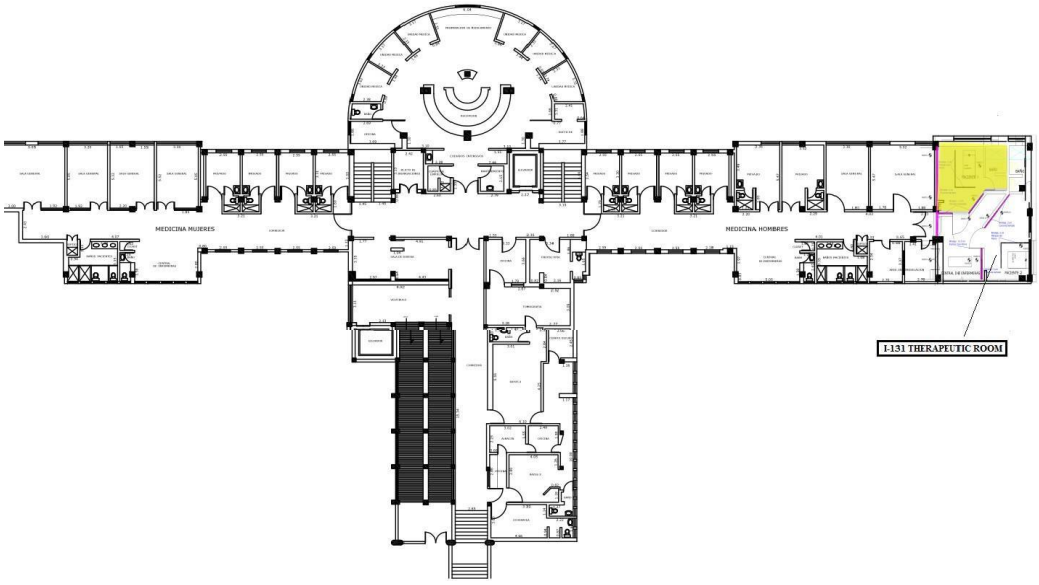

FIG. 1. Third floor of HUS with the indication of I-131 Therapeutic Room

It has to be mentioned that, as an old construction area, there is no information on the material used for their construction, as well as their thickness. There was also the need to use such structure with minimum of changes for not invading other hospital areas at low cost.

First, an I-131 punctual source was used to measure activity in different points through the walls. The I-131 source came from a supplier of the hospital, with 1.1729GBq (31.7mCi) in a lead container, used to simulate a punctual source. Blank (no shielding) activity measurements were taken at 1 meter distance from the radiation source, as can be seen in Fig. 2. Ten measures were taken and the media was calculated. The value founded for this source without attenuation was 6.00mR/h.

FIG. 2. Arrangement to measure the point source

The second step consisted in taking measurements ten times at each of the six different points through the walls, through the three existing doors and through the floor (on roof of second floor) and over the room roof. The measures were taken 0.30m from the walls, 1.70m above the lower floor and 0.50m over the upper floor [2] for calculating the acceptable doses. In the second floor under the therapeutic room, there is an unrestricted area, corresponding to the rest area for medical residents. The upper floor is a maintenance area. The decision to take the measures in different points was taken to evaluate homogeneity of the wall's material. Fig. 3 shows the walls and doors, and lower and upper floor.

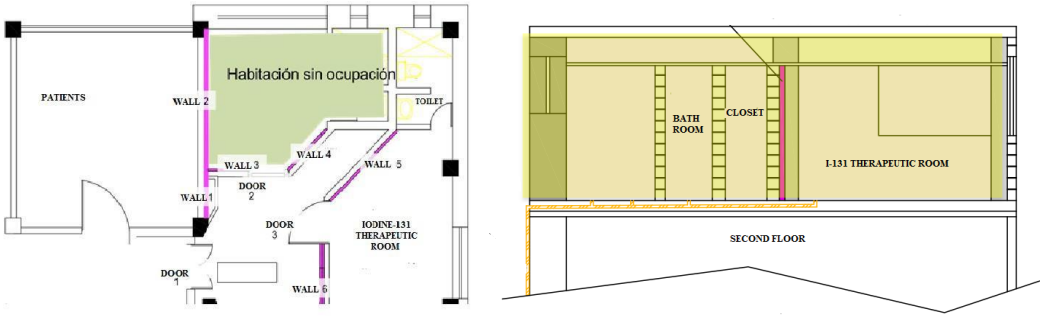

FIG. 3. Arrangement to measure the point source

The measuring points were decided as the ones with the higher activity measurements, which in turn were detected after a wall scan with the Geiger-Muller detector, taken one meter from the floor, which is in accordance with the patient's position on the bed (height of bed).

The activity measurements were taken with a Geiger-Muller Detector (GM) using the using 1 and 10 mR/h scales. The GM is described below:

- Brand: Ludlum.
- Model: 14C.
- Series Number: 236937.
- Detector Model: 44-38.
    - N/S: PR-251107.

Finally, after all measurements were taken, it was possible to calculate the attenuation, and of course, the absorption by the walls, doors, roof and the floor of the I-131 therapeutic room. The attenuation (A_t) was calculated by the equation (1):

$$A_t = 100\,[1 - (A/B)]$$

where:
A_t = attenuation (%)
A = measurement in different points (mR/h)
B = measurement without attenuation (mR/h)

And the absorption, is the difference from 100%, which represents the total radiation without the attenuation of the material.

The structural shielding calculations took into account the international effective dose limits, which is adopted from the publication ICRP 103 [4], and presented in Table 1.

TABLE 1. EFFECTIVE DOSE LIMITS

	Effective Dose Limits / year
Occupational	20 mSv
Public	1 mSv

[a] Average over defined periods of 5 years

These limits take into account the levels for occupationally exposed professionals (OEP) and for the other workers and public in general not involved with ionizing radiation (NO OEP). The effective dose limits considered are more restrictive than the ones coming from the General Radiological Safety Regulation in Mexico [5], which are in accordance with national rules.

The initial equivalent dose rate ($\dot{H}_0$) without the barriers were calculated for all measurement points (6-walls, 3-doors and upper and lower floor), following the equation (2):

$$\dot{H}_0 = \frac{A * \Gamma}{d^2}$$

where:

$\dot{H}_0$ = initial equivalent dose rate (mSv/h)

A = activity (MBq)

Γ = gamma constant (mSv.m^2/MBq.h)

d = the distance from the source to the point of interest

This calculation considered the activity of I-131 (7.4E+3 MBq) and the I-131 gamma constant (7.647E-05mSv.m^2/MBq.h [6]).

Moreover, it was also necessary to calculate the effective dose limits for both type of people (occupationally - OEP and non-occupationally exposed – NO OEP) in each surrounding area to the therapeutic room. It was necessary to analyze how much time the people remain in such areas, which depends if they work in hospital or only stay like a patient. Such time depends of many details but can be up to 2020h/year. The time is very important because they are going to expose less or more to such radiation. Such considerations are shown in Table 2.

TABLE 2. NEIGHBOURING AREAS TO I-131 THERAPEUTIC ROOM

Wall	Area	Controlled area	Personnel	Occupation (h/year)
1	general men's room	NO	NO OEP (patients)	120
			NO OEP (nurses)	720
2	general men's room	NO	NO OEP (patients)	120
			NO OEP (nurses)	720
3	closed room	NO	NO OEP	72
4	closed room	NO	OEP	72
5	corridor and closet	YES	OEP	1/5
6	nursing station	YES	OEP	2020
Door				
1	corridor	NO	NO OEP	1/8
2	closed room	NO	OEP	72
3	nursing station	NO	OEP	2020
Upper Floor				
1	maintenance	NO	NO OEP	1/40
Lower Floor				
1	medical residence	NO	NO OEP	882

3. RESULTS AND DISCUSION

The average measurements in the surrounding area to the **I-131 Therapeutic Room**, as well as the attenuation and transmission through the structural shielding, calculated using equation (1), are in Table 3.

TABLE 3. AVERAGE MEASUREMENTS, ABSORTION AND ATTENUATION BY THE STRUCTUTRAL SHIELDING

Wall (6 points)	Measurements (mR/h)	Transmission (%)	Attenuation (%)
without attenuation	6.00	###	###
1	0.25	4.17	95.83
2	0.22	3.67	96.33
3	0.30	5.00	95.00
4	0.50	8.33	91.67
5	1.00	16.67	83.33
6	1.00	16.67	83.33
Door			
1	2.20	36.67	63.33
2	2.00	33.33	66.67
3	2.00	33.33	66.67
Upper Floor			
1	0.20	3.33	96.67
Lower Floor			
1	0.20	3.33	96.67

Considering the results founded and presented in Tables 2 and 3, and applying the effective dose limits from Table 1, it was possible to calculate the necessity, or not, for increasing the structural shielding. And it was considered that for some positions there is the need of some extra shielding in the walls.

Taking into consideration the effective dose limits allowed for all kind of personnel in the surrounding area of the therapeutic room, all areas, except lower floor, are acceptable. Even for the nursing station, considered as non-occupationally exposed personnel, and considering their short exposition time, the acceptable effective dose limits is twenty times lower. However, for the lower floor, the need of such extra shielding was calculated to provide the effective radiation protection, as can be seen in Table 4.

TABLE 4. ADDITIONAL PROTECTION NECESSARY FOR THE LOWER FLOOR

Dose in area (mSv/year)	1 HVL[1] (3mmPb)	2 HVL (6mmPb)	Effective Dose Limit per year (mSv)
3.2	1.60	0.8	1

[1]HVL = Half Value Layer

4. CONCLUSIONS

- Transmission and attenuation of a gamma source were calculated by measurements through walls, doors and upper and lower floors of an I-131 therapeutic room.
- It is necessary to consider the kind of occupational areas and occupational time to evaluate the need of radiation protection for people surrounding the therapeutic room.
- It was necessary to add additional shielding material only for the lower floor, the additional material is lead with 6mm.
- The additional material on the floor is justified because the type and thickness of material used in the construction, along with the short distance between the patient and the users areas. The medical residents take their rest in bunk beds conducting to a very small distance between the radiation source, the patient, and the non-occupationally exposed worker – NO OEP.
- The costs is much lower than re-building with other shielding materials.
- This study allowed the director of the Institution for the decision to implement the new area.

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OUTPUT DOSE RATE IMPACT ON RADIOIODINE TREATED PATIENT-ACCOMPANIST IN EAST ALGERIA

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Abstract

Introduction:
The aim of this work is to evaluate the accumulated dose for accompanist (Da1) including journeys duration and patient-accompanist distances (d) inside a vehicle.

Methods and results:
Prospective work on 76 radioiodine treated patients; the output dose-rate at 1meter (Ḋ1m) is measured with a calibrated ionization chamber detector.
According to the different journeys duration and Ḋ1m at the exit: Da1≈ [0.9<15.65<60.1] microGy is under the exposure limit of 1mSv/year.
According to the variability of d for a patient with maximum Da1≈60.1 microGy in a journey of 3h:
- Da0.3≈667microGy for d=0.3m
- Da0.8≈93microGy for d=0.8m

The calculation of Da imposes Exposure Time Limits (ETL) to attend 1mSv, for the maximum D1m patient, for:
- d=0.3m a ETL=4.30h
- d=0.8m a ETL=32h
- d=1m a ETL=50h

Conclusion:
Nuclear medicine hospitalization allows us to protect public and environment. Only output dose-rate does not adequately evaluate the patient-accompanist exposure. In fact, duration of the journey and patient-accompanist distance can widely modify the received doses and requires special radiation protection measures for high D1m patients.

Topic: Radiation protection in nuclear medicine

1. INTRODUCTION

The main purpose of the hospitalization in Nuclear Medicine is to allow patient's circulation with a dose rate sufficiently low to a minimal irradiation of the public and the environment.
In this study, we will check the compliance of the exit dose rate according to the national obligation [1,2] and the international recommendations [3, 4,5,6]. Then, we will calculate the cumulative dose received by the accompanist of the hospitalized patient including travel time Finally, we will study the accompanist's variability of the cumulative dose according to patient-accompanist distance during travel time inside the vehicle for a particular patient(1).

2. PATIENTS AND METHODS

Prospective work on the radiation protection of the accompanist in the therapy unit, Nuclear Medicine Department, University Hospital Dr BENBADIS, Constantine, Algeria. Patients are hospitalized for an ablative radioiodine treatment using Iodine 131 for differentiated thyroid carcinoma. The principal goals are:

- To destroy any remaining tissue (normal or cancer residue) which facilitate future medical care by thyroglobulin follow up and any efficient uptake radioiodine metastasis.
- To allows post therapy total body scans.

76 patients received I-131ablative activities (2.9GBq-3.7GBq) are hospitalized for 3 days.

a- Dose rate (Ḋ) is measured daily using a calibrated ionization chamber detector (Victoreen 451B).
b- We calculate the cumulative dose received by patient accompanist from different dose rates (1 meter) and travel times (h) for each patient, as demonstrated in formula 1:

$$\boldsymbol{D} = \dot{D}_1(\blacksquare\blacksquare\blacksquare h) \times \boldsymbol{t}(h) \quad (1)$$

***$\dot{D}_1$*:** The measured dose rate at one meter.
***D*:** the cumulative dose received by the accompanist.
***T*:** the travel time.

c- The calculated cumulated dose at different patient-accompanist distances in the vehicle, formula 2(2):

$$\boldsymbol{D}_1 \times (\boldsymbol{d}_1)^2 = \boldsymbol{D}_2 \times (\boldsymbol{d}_2)^2 (2)$$

***D_1*:** The cumulated dose at one meter
D_2*:** The cumulated dose deducted at ***d_2
***d_1*:** patient-accompanist distance =1metre
***d_2*:** different patient-accompanist distances

d- Exposure time limit(4) was calculated from formula 1 at different patient-accompanist distance for a particular patient, we suppose that the exit dose rate do not change over the time (for few hours)

3. RESULTS AND DISCUSSIONS

a) Out of 76 patients the discharge dose rate was satisfying. Table 1 illustrates the results.

TABLE 1. PATIENTS EXIT DOSE RATE RANGE

values	minimum	Mean value	maximum	National obligation(3)
Dose rate(micro Gy.h^{-1})	0.9	7.5	20.02	37

(1) Particular patient: Maximum dose rate (20.02microGy.h^{-1}) and the longest travel time (3h).
(2) Formula 2: Based on Newton's inverse square law formula.
(3) Local work: Average Dose rate corresponding to the national obligation residual activity 1.11GBq (iode-131) Based on [5] (5microSv.h^{-1}≈150MBq).

(4) Exposure time limit: is time to reach a cumulated dose of 1milliGy corresponding to 1milliSv/year (for Iode-131) for a particular patient with 20.02milliGy.h^{-1} dose rate.

b) The cumulative dose received by the 76 patient accompanists, according to their respective travel time and the exit dose rate table 2, gives more than satisfying results.

TABLE 2. PATIENT'S ACCOMPANIST CUMULATED DOSE

value	minimum	Mean value	maximum
D (microGy)	0.9	15.6	60.1

c) Variation of cumulative dose in function of patient-accompanist distances: Considering
- Estimated cumulative dose for particular patient (1) is ***D*** =60.1microGy.
- Possible different patient-accompanist distances in the vehicle.

Figure1, shows that the accompanist's particular patient can reach a cumulated dose of 1000microGy (corresponding to the limit exposure of 1milliSv/year) in a single trip for 0.25 meter patient-accompanist distance.

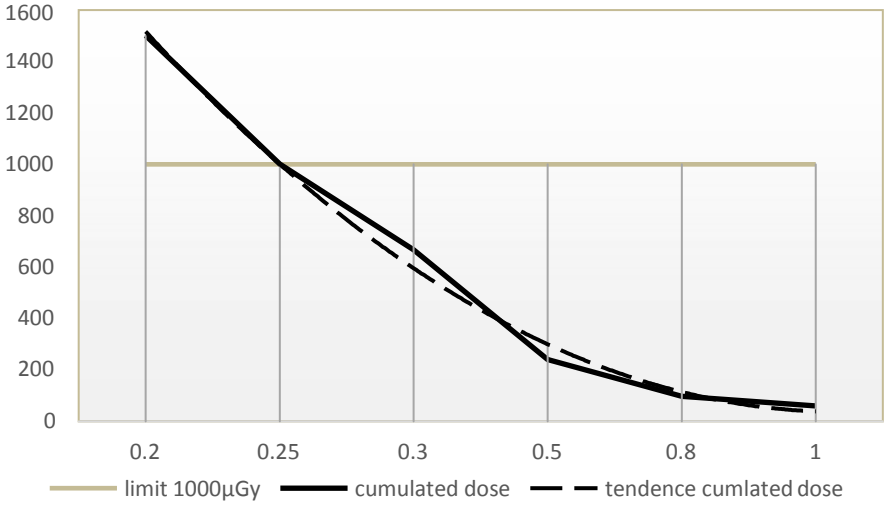

FIG.1. Variation of cumulated dose in function of patient-accompanist distance.

d) The calculations of the accompanist exposure time limit at different distance in the vehicle to reach a cumulative dose limit of 1mSv/year [1, 2], for a particular patient(1)are summarized in Table 3.

TABLE 3. EXPOSURE TIME LIMIT AT DIFFRENT PATIENT-ACCOMPANIST DISTANCES

Dose rate (microGy.h^{-1})	Distance (meter)	Exposure time limit(h)
20.02	0.2	2
20.02	0.3	4:20
20.02	0.5	12:30
20.02	0.8	32
20.02	1	50

4. CONCLUSION

Nuclear medicine hospitalization allows us to protect public and environment. Only output dose rate does not adequately evaluate the patient-accompanist exposure. In fact, journey duration associated to patient-accompanist distance can widely modify the received doses by the entourage and give results close to the regulatory limits; therefore these two parameters must be taken into account in the framework of an effective and personalized radiation protection.

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RADIOIODINE HOSPITALIZATION DURATION AT THE NUCLEAR MEDICINE DEPARTMENT CONSTANTINE ALGERIA

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Abstract

Introduction:

The goals of this study are:

- To check the compatibility of dose-rate at 1metre (D_{1m}) with national obligation and international recommendations.
- To study the possibility of hospitalization duration rearranging.

Methods and results:

76patients treated in our therapy unit for radioiodine ablative activities.D_{1m} is Measured daily (D_{1m24h};D_{1m48};D_{1m72h}) with a calibrated ionization chamber detector.

- The output D_{1m72h}≈[0.9<7.2<20] microGy/h is under the nationalobligation and international recommendations.
- The modulation in favor of reducing the length hospitalization isnecessary.
- The D_{1m48h} ≈[4.9 <18.6<40.7] microGy/h and D_{1m60h}≈[2 <11.4 <28.2] microGy/h are both acceptable but more advantageous in favor of theD_{1m}60h.

Conclusion:

In light of these results theD1m72h are in favor of decreasing length hospitalization ranging between 48h and 60h instead of the current 72h, where the need for a customized length hospitalization and radiation protection.

Topic: Radiation protection in nuclear medicine

1. INTRODUCTION

Given the exponentially increasing number of patients, we sought to find solutions to treat the maximum number of patients per week; we carried out studies and analyze in order to arrive at a possible rearrangement of the hospitalization duration of each patient and the goals were:

- To check the compatibility of exit dose rate at 1metre (D_{1m}) with national obligation [1, 2] and international recommendations [3, 4, 5].
- To study the possibility of hospitalization duration rearranging.

2. METHODS

It is a prospective study on 58 female and 18 male, treated in our therapy unit for radioiodine therapy with administrated activities (2.9-3.7GBq) during 3 days. $\dot{D}_{1m}$ is Measured daily ($\dot{D}_{0h}$; $\dot{D}_{24h}$; $\dot{D}_{48h}$; $\dot{D}_{72h}$) with a calibrated ionization chamber detector (Victoreen451B)

a. Given that the Algerian legislation [1] allows the exit of the patients treated for an iodine-131 therapy at a residual activity (A) less than 1.1GBq, the estimate dose rate corresponding to this one is approximately 37microGy.h^{-1}.according to formula 1 given by international recommendation [4]:

$$\dot{D} = 5\text{microSv.h}^{-1} \rightarrow A = 150\text{MBq} \quad (1)$$

The residual activities (A_{72h}; A_{48h}) corresponding to $\dot{D}_{72h}$;$\dot{D}_{48h}$ are also estimated by formula 1.

b. The dose rate at 60h was calculated after the estimation of the effective period for patients using MICROSOFT EXCLE application.
The corresponding residual activity (A_{60h}) is estimated by formula 1.

3. RESULTS AND DISCUSSIONS

a- The exit dose rates and their equivalent residual activities are summarized in table 1

TABLE 1. PATIENTS EXIT DOSE RATE AND EQUIVALENT RISIDUAL ACTIVITY RANGE FOR 72h

values	minimum	Mean value	maximum	National obligation
Dose rate (microGy.h^{-1})	0.9	7.5	20.02	37
Equivalent activity(MBq)	27	225	600.6	1110

Seen that $\dot{D}_{72h}$ and A_{72h} are under the national obligation and international recommendation, we tried to check $\dot{D}_{48h}$ and their A_{48h}. Table 2 gives the results.

TABLE 2. PATIENTS EXIT DOSE RATE AND EQUIVALENT ACTIVITY RANGE FOR 48h

values	minimum	Mean value	maximum
Dose rate (microGy.h^{-1})	4.9	18.6	40.7
Equivalent activity (MBq)	147	558	1221

Dose rates at 48h and their A_{48h} are acceptable for 95% of patient, so the results are acceptable but do not cover all the patients.

b- The dose rate at 60h and their corresponding activities are more acceptable. (100% of patients are under the national obligation [1] and international recommendations) [3, 4, 5]. Table3.

TABLE 3. CALCULATED PATIENT EXIT DOSERATE AND EQUIVALENT RESUDUAL ACTIVITY RANGE FOR 60h.

values	minimum	Mean value	maximum
Dose rate (microGy.h^{-1})	2	11.4	28.2
Equivalent activity (MBq)	60	342	846

All the dose rates at 72h, 60h, 48h are under the national obligation and international recommendation. Figure 1 summarized.

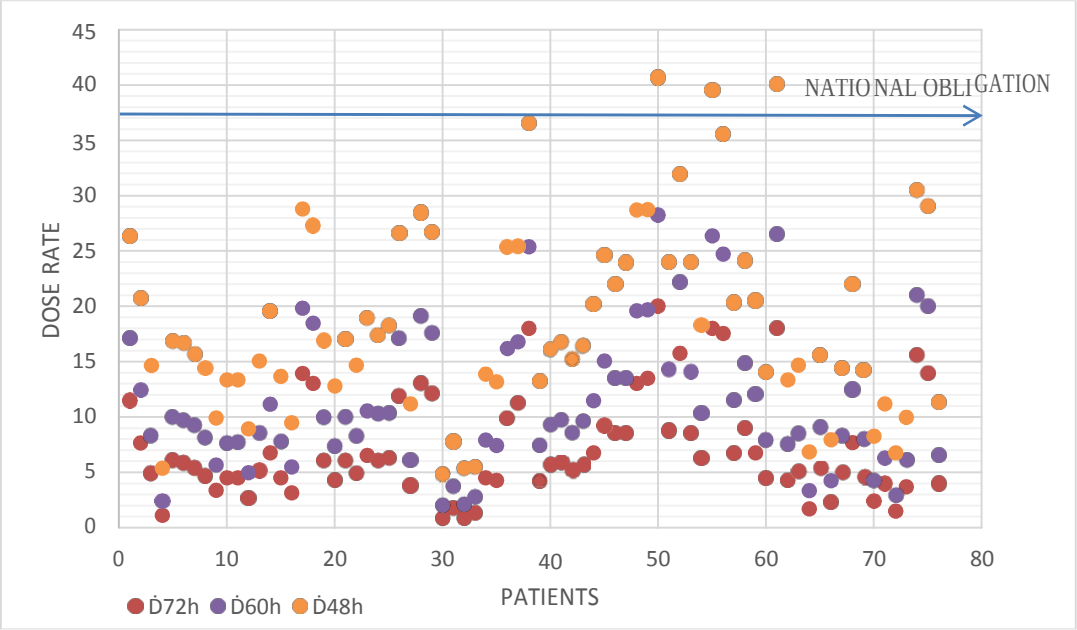

FIG.1. DOSE RATES AT 72H.60H.48H IN FONCTION OF DIFFERENT PATIENTS.

4. CONCLUSIONS

In light of these results the exit dose rate at 72 hours are in favor of decreasing length hospitalization ranging between 48h and 60h instead of the current 72h, where the need for a customized length hospitalization and radiation protection.

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PRECAUTIONS AND DOSE ESTIMATIONS FOR STAFF INVOLVED IN PET-CT TECHNOLOGY WITH ^{18}F-FDG

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Abstract

The paper brings up the questions of radiation exposure to personnel and appropriate radiation safety aspects in PET-CT department with own radiopharmaceutical production technology. The main sources of radiation exposure and the efficiency of applied shielding were analyzed. The dose rates on working places were studied using gamma, beta and neutron dosimeters and the effective doses to personnel, performing tracer production and PET-CT examinations were measured using electronic personnel dosimeters EPD Mk2 and thermoluminescence dosimeters TLD Harshaw 100. The obtained results showed good agreement between TLD and EPD measurements and the results of other authors. The maximum value of effective dose to worker appeared to be 1.65±0.25 mSv/year.

1. INTRODUCTION

PET-CT is a widespread method of diagnostics the usage of which leads to contact with radioactive sources. Technical and medical workers, involved in PET-CT technology undergo radiation exposure almost in all steps of its performance, such as tracer production, chemistry, dose preparation, injections, scanning etc. In this connection the radiation shielding and personnel monitoring during examinations and technical support of equipment is an important part of radiation safety requirements. The effective doses determination is necessary for the prevention of dose limits excess and plays a leading role in the adjustment of radiation reference levels. Besides, the analysis of the dose values allows to make a conclusion about achieved radiation safety conditions and implementation of new safe measures, directed at a minimization of radiation exposure to personnel.

The PET-CT technology with combined Biograph 64 scanner and self-shielded 11-MeV cyclotron Eclipse RD (Siemens) has been successfully implemented in All-Ukrainian Center for Radiosurgery of the Clinical Hospital "Feofaniya" in 2012 [1,2]. It was the first experience of PET-CT performance in Ukraine, and therefore special attention has been paid to the radiation safety aspects, starting with the rooms design and up to measuring equipment. The technical and organizational measures have been provided in order to minimize the additional exposure to personnel according to the ALARA principle.

2. SOURCES OF EXPOSURE AND SHIELDING

There are following steps of PET-CT performance mixed with radiation exposure to employees: tracer production, dose preparation, injection, patient positioning and patient escorting [3]. Each of these steps is done by its own responsible worker and Table 1 shows the step and the person, that executes corresponding step. For dose rate reduction these steps are carried out using different types of protection, such as distance, time or physical barriers.

Tracer production technology includes service and technical support of a cyclotron, hence cyclotron exploitation is accompanied by several sources of radiation that are: prompt gamma-rays and neutrons, produced tracer and induced activity of a cyclotron components. The prompt radiation exists only during irradiation and the bigger part of it is absorbed by the cyclotron self-shielding, consisting of concrete, borated polyethylene and lead. After bombardment and tracer delivery to the lab, the relicts of produced tracer and induced activity

become as the main sources of radiation. The highest activation occurs near the target module, including vacuum window, Havar foil, collimator (spacer) and target body components.

After tracer production, synthesis and quality control of radiopharmaceutical it goes to dose preparation room, where the measurement of FDG activity is carried out with subsequent injection to the patient. Hence, the next main sources of radiation are contaminated medical tools and the patient. In the first case radiation consists of annihilation photons and beta-particles, while in the second one the greater part of positrons is absorbed by patient and additional exposure to staff is mostly caused by 511-keV gamma-rays. For dose reduction during injection the 9 mm tungsten shielding for syringe is used and then the contaminated tools are placed in lead containers for further decay. During patient management, in its turn, time and distance are the main means of protection.

It should be noted that apart from diagnostics examination the physicist also does the quality control of PET equipment with ^{68}Ge, ^{133}Ba, ^{137}Cs, ^{60}Co and ^{57}Co isotopes in the form of enclosed calibration sources and that also contribute some part of individual dose to physicist involved in PET-CT technology. The calibration sources are kept in lead containers with corresponding marking giving information about nuclide, activity and source number.

TABLE 1. THE STEPS MIXED WITH EXPOSURE IN PET-CT TECHNOLOGY

Step	Protection	Responsible person
Tracer production	Cyclotron self-shielding	cyclotron engineer
Dose preparation	Fume hood, leadglass window and lead shielding	nurse, physicist
Injection	Tungsten outer cover for syringe	nurse
Patient positioning	Time, distance	lab-assistant
Patient escorting	Time, distance	lab-assistant

3. EQUIVALENT DOSE RATE MEASUREMENTS

The each way of potential exposure to personnel was analyzed by equivalent dose rate measurements on working places using corresponding detectors, sensitive to gamma, beta (RadEye B20 [4]) and neutron radiation (RadEye Px with NRD probe [5]).

Concerning tracer production, the prompt gammas and neutrons exist only during irradiation and they are almost fully absorbed by the cyclotron self-shielding. The dose rate on shielding surface during irradiation is 1 mSv/h for neutrons and 75 μSv/h for gamma rays respectively. The working place is located in the separate control room, where dose rate does not exceed 0.35 μSv/h. At the end of bombardment, the radiation from induced activity is fully absorbed by self-shielding, but when the shield is open the dose rate near the target module can be very high (more than 15 mSv/h), and for technical maintenance it is necessary to wait at least for a one day for short-lived radionuclide decay. The measured dose rates during cyclotron operation and maintenance, including working places and target module components are presented in Table 2. According to obtained dose rates it is clear, that the bigger part of dose to cyclotron engineer is contributed by technical support of target module, as in the other departments, where the same cyclotron is used [6].

The highest dose rates in PET-CT examination were measured during injection (≈300 μSv/h) and patient positioning (≈200 μSv/h), while from calibration sources dose rate varies from 0.5 μSv/h and up to 1 mSv/h (for 70MBq of ^{68}Ge).

TABLE 2. THE MEASURED DOSE RATES DURING CYCLOTRON EXPLOITATION

Point		Measurement conditions	Dose rate, mSv/h
Control room		During bombardment	$0.35 \cdot 10^{-3}$
Shielding surface		During bombardment	1.075
Target module	Target body	Usage: 5210 μA·h Time after EOB: 60 h	0.14
	Collimator (spacer)	Usage: 5210 μA·h Time after EOB: 60 h	0.05
	Havar foil	Usage: 1950 μA·h Time after EOB: 60 h	5.42
	Vacuum window	Usage: 3540 μA·h Time after EOB: 60 h	2.52

4. INDIVIDUAL DOSES AND DATA ANALYSIS

The average effective dose to employer per year depends on a workload of the department, that can be expressed by the number of production runs for cyclotron engineer and the number of PET-CT examinations for nurse, lab-assistant and physicist. The effective doses to personnel were measured using electronic personnel dosimeters EPD Mk2 [7] and thermoluminescence dosimeters TLD Harshaw 100 [8]. The EPD measurements were performed for specified step, while TLD measurements were carried out whole working time. The measurements were performed during 2016 (there were 219 production runs and 1098 PET-CT examinations in this period) and the obtained results are presented in Table3.

As it was mentioned above, during cyclotron exploitation the bigger dose to engineer goes from service and technical support of facility, mainly of target module. The average target lifetime is about 1950 μAh [9] (in our Center it is reached after 3 months), so there are minimum 4 target replacement actions per year, excluding periodic target leakages and blowing of vacuum window. Moreover, there is another repair actions, to be done near the target module, such as extractors, ion source cathodes and vacuum window replacements. The average effective dose to cyclotron engineer was determined as a sum of EPD measurements for different actions and with the help of continuous TLD measurements, giving the values of 1.054 mSv/y and 1.08±0.15 mSv/y respectively.

The doses from PET-CT examinations were measured using EPD for one PET-CT study and with the help of TLD measurements for whole working time. The lab-assistant and nurse receive higher doses per one study in comparison with physicist, but taking into account the number of occupied workers the average dose per year is close for each of them (from 1.28 mSv/y for physicist and up to 1.65 mSv/y for lab-assistant). A lot of other PET Centers use only one person (technologist) for all steps execution and the dose to this worker equals about 7÷10 μSv/study [10-12], that is in a good agreement with our results (in our case the sum dose is 9.42 μSv/study).

In general the TLD measurements give higher values of doses than a sum of EPD measurements because of the additional exposure from other ways, caused by working environment, quality control procedures, calibration sources and FDG management.

TABLE 3. THE DOSES TO WORKERS MEASURED USING EPD AND TLD DOSIMETERS DURING 2016

Action (the number of actions performed per year)		Responsible person (the number of occupied workers)	Dose/action, μSv (EPD)	Average dose to worker, mSv/year (TLD)
Cyclotron service	Target rebuild (5)	Cyclotron engineer (2)	152	1.08±0.15
	Extractor replacement (1)		75	
	Cathodes replacement (1)		63	
	Vacuum window replacement (2)		78	
			Sum 1054	
PET-CT examination (1098)		Nurse (4)	4.31	1.41±0.06
		Physicist (2)	1.82	1.28±0.10
		Lab-assistant (3)	3.29	1.65±0.25
			Sum 9.42	

5. CONCLUSIONS

The potential ways of radiation exposure to PET-CT staff and corresponding radiation protection were analyzed. The working environment was studied by dose rate measurements and effective individual doses determination by EPD and TLD dosimeters.

For the tracer production technology the most intensive exposure (up to 5-6 mSv/h) is observed during target module management and corresponding repair actions. But the low number of this actions per year ensures the effective dose to cyclotron engineer to be about 0.5 mSv/year (1.054 mSv/y divided by 2 engineers). The dose rate at working place during production is near the background (0.35 μSv/h) and even 219 production runs do not make significant contribution to the effective dose. The dose to cyclotron engineer, measured using TLD dosimeters is equal to 1.08±0.15 mSv/y.

For PET-CT personnel, the highest doses were measured for lab-assistants, whose additional exposure is caused by patient management. However, even almost 1100 PET-CT examinations per year deliver the dose not more than 1.65±0.25 mSv/y. The sum dose per one PET-CT study (9.42 μSv) is in the good agreement with the results of other authors.

The measured effective doses appeared to be considerably lower than adopted dose limits, recommended by International Commission on Radiological Protection [13] and the maximum dose reached the value of 1.65±0.25 mSv/y. The relatively low effective doses in comparison with prescribed dose limits (max. 9.5% of the 20 mSv/y limit) prove good radiation safety margin and safe measures, directed at radiation safetyculture.

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PRODUCTION OF SYNTHETIC CRYSTAL OF $CASIO_3$ AND ITS APPLICATION IN LOW-DOSE DOSIMETRY RADIATION

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Abstract

The use of different types of radiation is being increasingly widespread in various human activities such as intensive development radiation technologies, high nuclear energy, and nuclear medicine. The radiation, moreover, can be harmful to humans and dosimetry radiation becomes very important. There exist different systems in dosimetry radiation, based on thermoluminescent crystals being the most important one. Was produced synthetic polycrystals of calcium silicate $CaSiO_3$ by devitrification method, which in addition to the low cost and easy production it proved to be very sensitive as far as TL is concerned, in another word, it can be a useful radiation dosimeter. It has shown that 20 mg of $CaSiO_3$ can detect X-ray produced by 6 MeV electrons from a linear accelerator at a private Sirio-Libanés hospital in São Paulo, for its use in radiotherapy with a dose test of 1 Gy. In addition, 5.6 mg of $CaSiO_3$ in grains of 75 – 180 μm size was exposed to 662 keV gamma rays from a Cs-137 source with a dose test of 10, 20 and 30 mGy, being sensitive at these doses. Therefore, there is a chance that it detects doses as low as few mGy such that it can be used in nuclear medicine or monitoring actual radiotherapy treatment.

1. INTRODUCTION

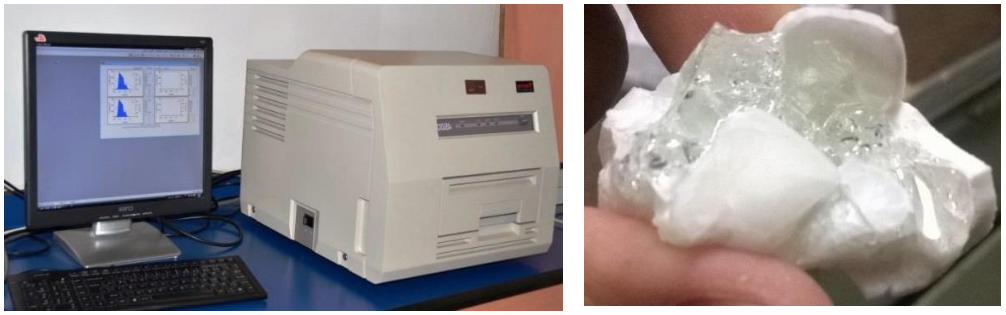
Different types of radiation are being increasingly widespread in nuclear medicine, in radiotherapy, etc., very low dose radiation is involved. Besides such usefulness of radiation, it can also induce some harm to the health of human involved with radiation. Hence the radiation dosimetry and establishing rules to protect against radiation become of most importance.

The dosimetry based in thermoluminescence (TL), the so-called TLD, started in late 1950's. LiF doped with Mg and Ti and later with Mg, Cu, P have been and still they are used as dosimetry materials [1]. Several other ionic crystals have been studied as dosimetry materials, among them $Al2O_3$: C and some others are also used extensively. At the Ionic Crystal Laboratory of the University of São Paulo, natural silicate minerals have been the object of investigation concerning their color, centers, Electron Paramagnetic Resonance (EPR) and TL properties, particularly under radiation. Many of these minerals have shown a high TL sensitivity, especially for high, sometimes, very high radiation doses. Some variety of quartz beryl, tourmaline, jadeite and other minerals can thus be used to monitor high-dose irradiation for food preservation, the color change of previous stone and other industrial applications, very often million Gy radiation are used. On the other hand for medical applications very low dose radiation are used. Organic materials are often used for dosimetry, however, only very few natural silicate minerals respond to such radiation, such as green quartz. In the present work, we produced synthetical silicate minerals to see its behavior under low and high dose irradiation.

2. MATERIALS AND METHODS

To begin with, we aimed to produce a synthetic $CaSiO_3$. Starting with the stoichiometric balance of the components that were used. The chemistry equation used is: $CaO + SiO_2 \rightarrow CaSiO_3$, theoretical composition: CaO = 48.3% and SiO_2 = 51.7%, but for to ensure the production of the Calcium metasilicate, we use: CaO = 40.0% and SiO_2= 50.0% [2]. Thoroughly mixed in an alumina crucible this mixture is melted at 1550° C for two hours and then the temperature is lowered in a controlled manner by about 48 hours (devitrification method) [3]. Polycrystals of calcium were obtained and since the grain sizes are variable, they are sieved in such a way that grains of 0.080 to 0.180 mm size are retained for TL measurements.

These were carried out in Harshaw 4500 TL reader keeping 4° C/s heating rate. The irradiation was carried out in ^{60}Co γ-ray source at the Institute for Energy and Nuclear Researches (IPEN) in São Paulo, Brazil. Figure 1, (a) shows the Harshaw 4500 equipment and (b) calcium silicate polycrystalline obtained.

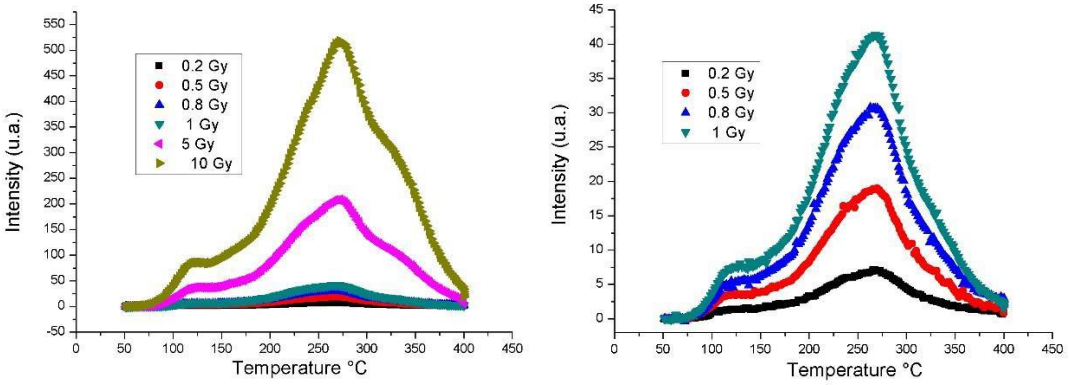


(a) (b)

FIG.1. (a) Harshaw 4500 equipment, (b) synthetic $CaSiO_3$ polycrystalline.

3. RESULTS AND DISCUSSIONS

Figure 2, shows glow curves for doses from 0.2 to 10 Gy using about 5.6 mg mass for each reading. One prominent peak around 120 and 260° C, are observed. Also, a shoulder can be seen around 320° C. Figure 3, shows TL response as a function of doses, the growth curve indicates the possibility to detect doses lower than 1.2 Gy. Actually, the TL response grows linearly up to about 1000 Gy and sub-linearly for doses above this value.



(a) (b)

FIG. 2. TL glow curves of synthetic $CaSiO_3$ using a reading mass of 5.6 mg. (a) from 0.2 to 10 Gy, and (b) 0.2 to 1 Gy.

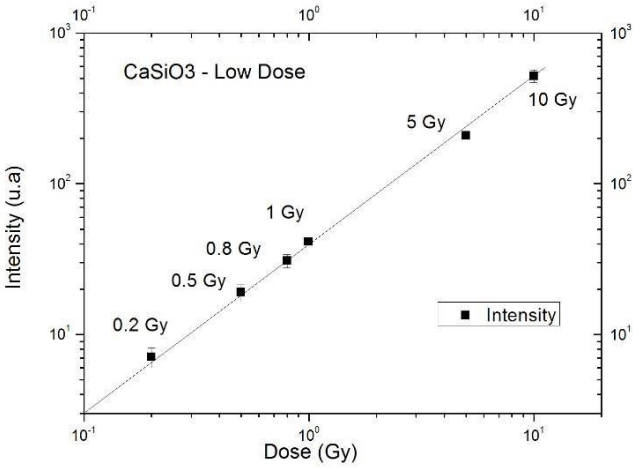


FIG. 3. TL response of synthetic $CaSiO_3$ as a function of doses.

Polycrystals of $CaSiO_3$ in grains of 75 -180 µm size, were exposed to 662 keV γ-ray from a Cs-137 source at IPEN. Figure 4, (a) shows the TL glow curves of $CaSiO_3$ irradiated with γ-ray doses of 10 mGy, 20 mGy and 30 mGy using a reading mass of ~5.6 mg, and (b) shows glow curves of $CaSiO_3$ irradiated with the same γ-ray doses using a reading mass of ~9.0 mg.

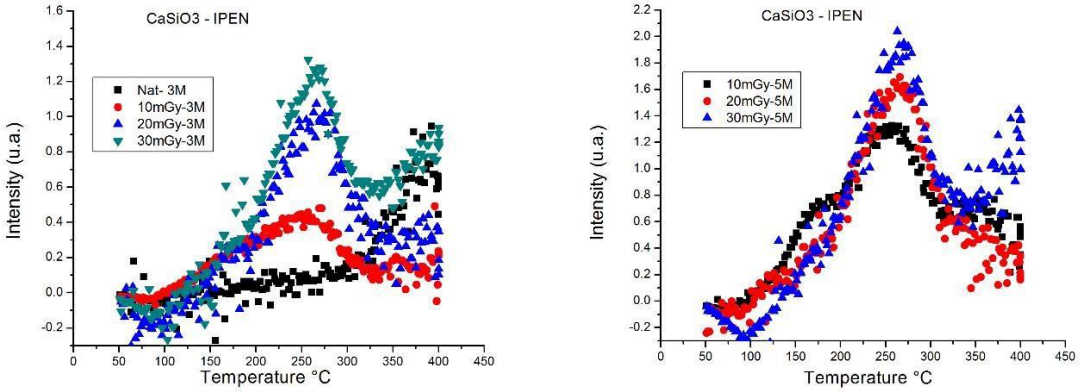


FIG. 4. Glow curves of synthetic $CaSiO_3$ irradiated with γ-ray doses of 10 mGy, 20 mGy and 30 mGy using a reading mass of 5.6 mg; (b) irradiated with the same doses using a reading mass of 9.0 mg.

Figure 5, shows TL as a function of doses around the prominent peak around 270° C, including the TL peaks of 10 mGy, 20 mGy, and 30 mGy. In all cases was used a reading mass of ~ 5.6 mg.

In addition, $CaSiO_3$ was exposed in grains to 6 MeV X-ray from a linear accelerator at a private Sirio-Libanés hospital in São Paulo, for its use in radiotherapy. The dose deposited of the equipment was 0.97136 ± 0.0194 Gy and calculated by Calcium Silicate Polycrystalline was 1.246 ± 0.046 Gy, obtaining a relative error of ~ 28 %. The mass used was 20 mg.

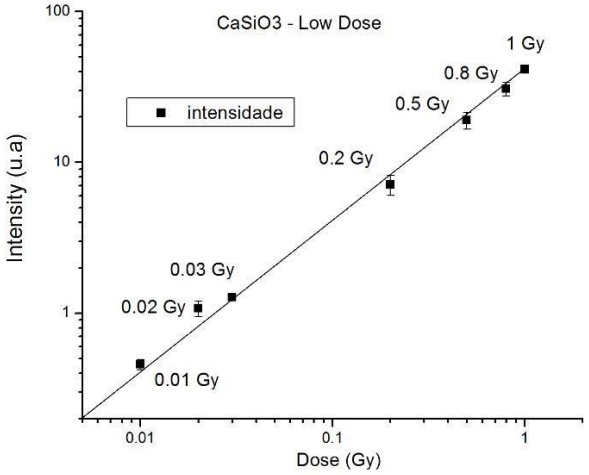


FIG.5: TL as a function of doses of synthetic $CaSiO_3$ around the prominent peak around 270° C including the TL peaks of 10 mGy, 20 mGy, and 30 mGy.

4. CONCLUSIONS

The behavior of the glow curve TL of the synthetic CaSiO3 at low dose makes it a strong candidate for use in medical applications. The next step will be the production of chips for a good handling to the synthetic silicate in future analysis.

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SAFETY FOR M^{131}IBG TREATMENT WITH PERISTALTIC INFUSION PUMP PROCEDURE

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Abstract

^{131}I plays an important role for therapeutic in Nuclear Medicine. However, radiation hazard of ^{131}I is crucial not only from external radiation but also internal dose due to its volatile property. m^{131}IBG is one of the radiotracers to treat neuroblastomas by administration m^{131}IBG solution to patients. Aim of the work was to access the radiation safety of the treatment procedure. **Method:** m^{131}IBG was administrated by using peristaltic infusion pump. In the line of administration, the air venting unit with or without a charcoal column was used. During the treatment with 5.5 GBq m^{131}IBG, radiation exposure to the staff and air contamination with ^{131}I was determined with pocket dosimeter and air sampling pump, respectively. **Results**: The radiation doses to the staff were 19 and 16 μSv during the treatment periods. Derived air concentration (DAC) of ^{131}I in the treatment room was 20,280 Bq/m^3 and 170 Bq/m^3 when air venting unit without and with charcoal column was used, respectively. **Conclusion:** The external radiation exposure was less than the occupational radiation dose limit whereas ^{131}I in air was only lower than the DAC limit when charcoal column was applied. Therefore, the procedure was safe and suitable for routine service.

1. INTRODUCTION

^{131}I has been commonly used and plays important role in Nuclear Medicine especially for the therapeutic purpose. However, its hazard is possibly not only from external radiation exposure but especially also from internal radiation dose due to volatility property of ^{131}I [1]. Therefore, radiation hazard of ^{131}I is very crucial and needed to consider.

m^{131}IBG (^{131}I metaiodobenzylguanidine) is one of the radiotracers to treat neuroblasmtomas, phaeochromocytomas especially in paediatric patients. In general, ^{131}I-NaI (sodium iodide) for treating thyroid diseases can be prepared in the solution form or capsule form. However, only m^{131}IBG solution is available and suitable for the treatment and the free iodide of m^{131}IBG is possible up to 8 % [2]. As administrated activities of m^{131}IBG are 3.7- 5.5 GBq, the free iodine can be as high as 440 MBq. Additionally, m^{131}IBG solution is needed to infuse slowly to the patients around at least 1 h due to the adverse reaction (transient hypertension) [3]. These lead to increase the risk of inhalation of ^{131}I to the worker who deals with m^{131}IBG treatment procedure. Therefore, aim of this work was to access the radiation safety for this type of treatment.

2. MATERIALS AND METHODS

The treatment with m^{131}IBG was performed in the isolated room with the radiation protection equipment. Portable lead shield was utilised to protect external radiation from high gamma energy of ^{131}I by placing the shield between working officer and the patients including m^{131}IBG vial and all equipment for administration. The working officer who responded to the administration process wore digital pocket dosimeter. The radiation exposure was recorded after ending of the treatment.

To administration m^{131}IBG, peristaltic infusion pump was used to slowly infuse m^{131}IBG solution into the patients with the accurate infusion flow rate. In the infusion system, air venting unit which was 26-gauge needle attached with the 0.22 μm membrane filter was used. In addition, in-house charcoal column was packed with 2 g of activated charcoal filled into 5 ml syringe (as shown in Figure 1) and then attached to the air venting unit. Then this charcoal column was included in the following administrationprocedure.

Fig. 1. In-house charcoal column

Five point five gigabecquerel of m^{131}IBG solution (TINT, Thailand) in the vial was administrated to the paediatric patients. The line of infusion, air venting unit with or without charcoal filter was inserted into m^{131}IBG vial in order to demonstrate whether charcoal filter can diminish the level of ^{131}I contamination in the air. During the treatment, radiation exposure to the staff was determined with pocket dosimeter. For monitoring air contamination with ^{131}I, air sampling pump (with charcoal filters) was operated before starting the administration process for 45 min. Then the new filter was replaced the used one prior to start the treatment. The air sampling pump was operated with the second filter over the treatment period. All the charcoal filters were transferred to TINT for the following process. The charcoal filters were measured the radioiodine activities by using Gamma Spectroscopy HPGe detector and eventually subjected to calculate the DAC of ^{131}I at TINT to obtain the DACs at the rest and at the workingperiod.

3. RESULTS

The results of radiation doses to the staff measured with real time dosimeter were 19 μSv and 16 μSv during the treatment period of 2 patients for 2.5 h. The radioiodine activities were measured from the charcoal filters and analysed for the DACs of ^{131}I. The results showed that ^{131}I level in the air in the treatment room prior to operating (rest period) was lower than limit of detection (1.74 Bq/m^3) in the both cases. While during the treatment process (working period), the DAC of ^{131}I was 20,280 Bq/m^3 and 170 Bq/m^3 when air venting unit without and with charcoal filter column was used in the infusion system, respectively.

4. DISCUSSIONS

Apart from beta ray emission for therapeutic purpose of ^{131}I, it also emits high energy gamma radiation, 364 keV, with high abundance [4]. So radiation protection is necessarily considered especially in high dose treatment for thyroid carcinoma or neuroblastomas. As mentioned, m^{131}IBG is available in the solution form. In some centres, this radiotracer tracer is manually withdrawn from original vial into syringe and use infusion pump for processing the administration resulting in increasing the radiation exposure. In order to reduce the exposure from dispensing of high radiation activity, peristaltic infusion pump is chosen to replace the infusion pump. Then administration process can be done from the original m^{131}IBG vial that would diminish radiation exposure dose to the staff

The patients treated with high-dose (5.5 GBq) of m^{131}IBG are in-patient and separated in the radiaiodine treatment room. The radiation officers routinely monitor radiation exposure by either accumulated radiation dosimeter such as OSL (Optically stimulated luminescence) or TLD (thermoluminescent dosimeter) or real time dosimeter. In this work, digital pocket dosimeter was used to measure real time radiation exposure to the staff. The results show that the external radiation dose to the officer for the m^{131}IBG treatment with peristaltic infusion pump is around 16 to 19 μSv for one treatment (within treatment time of 2.5 h). Approximately 25 patients were undergone this treatment in 1 year. Also the staff has been rotated to perform the procedure. Therefore, theradiation exposure of the staff is lower than occupational radiation dose limit of 20 mSv/year.

Regarding volatile capability of ^{131}I at low pH condition, internal radiation exposure of ^{131}I [1] is another crucial issue to concern. There is possibility that free radioiodine in vapour form can be occurred in the m^{131}IBG solution (pH around 4.7). Because peristaltic infusion pump was used, the air venting unit is necessarily included in the treatment system. Then ^{131}I vapour might be able to freely exchange with the surrounding air in the treatment room via venting unit as shown that the concentration of ^{131}I in the air was 20,280 Bq/m^3 before attaching charcoal column in the system. That was obviously greater than DAC limit of ^{131}I of 416.67 Bq/m^3 [5]. In contrast, the level of ^{131}I in the air was 170 Bq/m^3 which was significantly lower than the DAC limit when charcoal column was applied. The result indicates that free radioiodine in the m^{131}IBG solution can escape from the vial via air venting unit resulting in exceeding the DAC limit of ^{131}I. However, in-house charcoal column is sufficient to prevent this phenomenon leading to decreasing of ^{131}I contaminate in the air to safetylevel.

5. CONCLUSION

The use of peristaltic infusion pump for m^{131}IBG treatment is the safe procedure for the staff from either external or internal radiation exposure. However, additional radiation protection device to efficiently absorb volatile ^{131}I such as charcoal column is essentially required to include into the administration system. Additionally, management of the waste of radioactive iodine solution should be performed with precaution and needed further investigation of the radiation safety for the staff, public and environment.

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Careful planning can reduce PET-CT room shielding

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Key words: PET/CT shielding

Abstract

Aim:
To calculate the necessary shielding requirements for a Philips Big Bore 16 slice, Positron Emission Tomography-Computed Tomography (PET-CT) scanner.

Introduction:
Radiation producing equipment needs to be shielded, usually with concrete or lead, to provide a radiation safe environment for the staff and general public close to the facility. Shielding of the equipment should limit radiation exposure to staff and the public to levels required by the Directorate: Radiation Control. The shielding requirements for a PET-CT unit involve both the PET and the CT components. While the PET part of the equipment does not produce radiation, the shielding has to shield adequately against the two 511 keV photons emitted from the injected radionuclide in the patient. The CT part of the equipment produces radiation that needs to beshielded.

Method:
When planning the new PET-CT installation and building at Tygerberg Hospital in the Western Cape, available literature and guidelines for PET-CT room shielding were reviewed. Some of the guidelines reviewed apply to small buildings only, and expensive overshielding can result if these guidelines are followed blindly. A comprehensive Excel sheet was prepared to take into account all the factors that contribute to the required shielding.

Results:
Less shielding could be used at our site because of the large room that houses the PET-CT scanner. From the calculations it was evident where to place the patient waiting rooms to reduce unnecessary shielding at the control room.

Conclusion:
Careful planning on the placement of the PET-CT scanner and the size of the room, taking into account the activity injected per patient and the workload, reduced the necessary shielding dramatically. This reduced cost and provided a radiation safe environment for staff to work in.

Introduction

A PET-CT scanner is a diagnostic imaging device detecting the biological uptake in a patient administered with a positron emitting radionuclide. The Computed Tomography (CT) component provides the attenuation correction for the PET images and identifies the anatomy of the patient relative to the PET uptake areas.

The CT scanner is radiation producing equipment that needs protective shielding when the scanner is busy scanning. The PET unit is an imaging device that detects radiation being emitted from the patient to form an image and has no need to be shielded. However, the patient injected with the radionuclide needs to be shielded. The dose rate constant of 0.147 µSv/MBq.h at 1 m applies to the positron emitting radionuclide F-18 as used in FDG. The body of the patient absorbs some of the annihilation photons and reduces the external dose rate by approximately 36 %.[1] The patient voiding will reduce the activity by another 15 – 20 % in the first two hours after the patient is injected and has to wait 45 - 90 min before being scanned. A typical workload of 10 patients/d x 5 d/week that equal 50 patients/week are applied for shielding calculations in the literature.[2] The "hot lab" where the manufactured radionuclide is delivered and where it is divided into patient specific doses also needs to be shielded adequately.

A Philips Gemini Time of flight Positron Emission Tomography-Computed Tomography (PET-CT) scanner, incorporating a Philips Brilliance Big Bore 16 slice Computed Tomography scanner, was acquired for Tygerberg Hospital in 2010. A large room that was adjacent to the oncology building was identified to house the PET-CT scanner. The initial room design needed to be adjusted and the medical physics division was appointed to supply the architect with the necessary wall thicknesses and other shielding requirements to make this a radiation safe area for staff and members of the public.

Method

Together with the architect it was decided on the layout of the department. Early on it was decided to split the department into two areas: one including the reception and reporting room and one including the hot lab, scanner and injected patient. From the floor plan areas were identified that would need shielding: Firstly from the radiation emitted by the CT scanner while in use, which included the waiting area, reporting room, injection area, patient quiet cubicles, hot lab, imaging room, control room as well as the outside of the building; Secondly areas where the patient would wait to be scanned and the time the patient would be in the PET-CT room.

Consultation with the clinical staff was conducted to establish the intended work load and the dose per patient (or per kg, including a minimum dose), scan time per patient (time the patient would spend in the PET-CT room), as well as the time each injected patient would spend in the building.

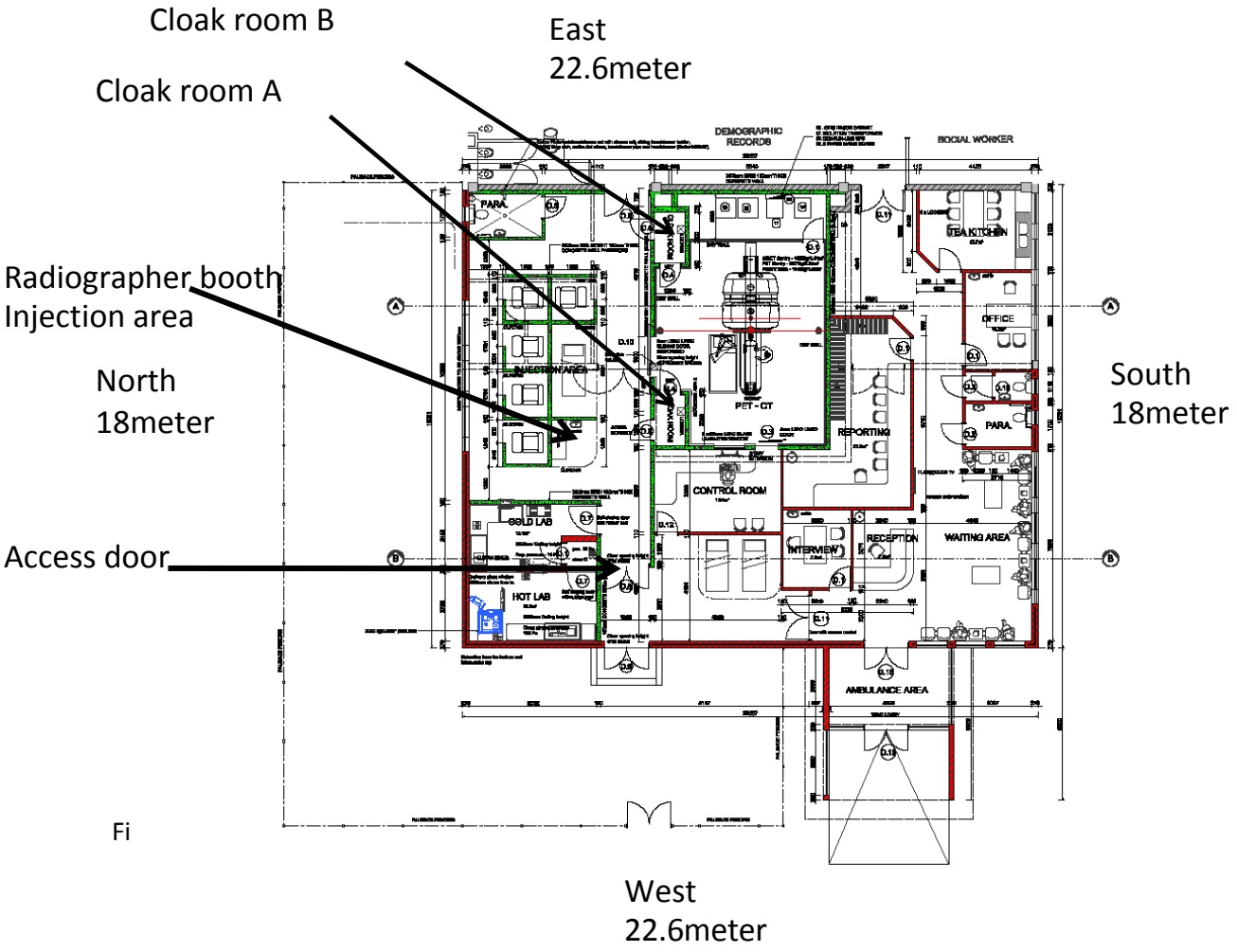

Fig 1: Initial floor plan housing the PET-CT scanner at Tygerberg hospital

Comprehensive MS Excel spreadsheets were prepared incorporating the workload, injected dose, CT radiation time on, staff presence in each area, public members in the area, distance from radiation and barrier factors to calculate the wall thickness for each area from the CT scanner, as well as from the radioactive patient.

Criteria that were applied for calculations included that the effective dose equivalent in uncontrolled areas should not exceed 1mSv/year (or 20 µSv/week) with an instantaneous exposure rate limit of 20 µSv in any 1 hour.[1] In controlled areas the occupational dose limit is 20 mSv/year. To be consistent with the ALARA principle, the target level of 5 mSv/year was used for calculations. This equates to 100 µSv per week. To apply these constraints, a distinction was made between controlled radiation areas and uncontrolled areas open to the public.[3]

Results

Main criteria used to reduce shielding requirements for our department included:

1. The amount of activity injected per patient at Tygerberg hospital is 2.8 MBq/kg, which equates to 195 MBq per 70 kg patient (but not less than 185 MBq per patient), which in turn is much less than the published calculations, that assume 555 MBq per patient.[1] Tygerberg's

PET scanner has Time of Flight capabilities which improves image quality, acceptable images can be produced for reporting with less injected activity.

2. 6 m length is adequate to house a PET-CT scanner, but we made use of an existing room of 9m length, which placed the machine further away from the control room. The additional length of 3 m reduced the exposure rate by a factor of 2.25, purely based on inverse square law calculations.

3. Patient throughput is less than with a normal diagnostic CT scanner. Ten patients are each receiving a 1 minute scan per day.

4. The patient waiting booth was moved further away (from 1 m to 8 m) from the control room, which results in a decreased exposure rate in the control room by a factor of 64.

The calculations from the Excel sheet showed that the patients waiting in cloak room A contributed to unnecessary radiation next to the control room, which would have necessitated a 4.9 cm thick concrete or 2.9 mm thick lead shielding wall. By moving the cloak room A to the end of the room next to cloak room B, no shielding was needed between these two areas.

FDG injected dose is less at our site and the size of the room that houses the PET-CT scanner is larger than usual resulted in a reduced wall thickness for the diagnostic room from 30 cm concrete to 15 cm concrete. No lead window was needed between the scanner room and the control area from a published 1.33cm Pb requirment[4]. Nonetheless a 2.2 mm lead window was installed corresponding to the minimum for X-ray units.

Conclusion

Careful planning on the placement of the PET-CT scanner and the size of the room, taking into account the activity injected per patient and the workload, reduced the necessary shielding dramatically. This reduced cost whilst providing a radiation safe environment for staff.

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Design and implementation of shielding to ensure radiological protection for patients, the public and the environment, adapted to the hybrid Nuclear Medicine SPECT / CT at IICS – UNA.

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INTRODUCTION

Concepted at the beginning of the 20th century, Radiation protection (RP) is a new discipline in modern science. The appearance of probable adverse effects in multidisciplinary professionals known as Operationally Exposed Workers (Trabajadores operacionalmente expuestos, TOES) arised later in the application. Also discovered, was the need to achieve the adequate levels of protection for people (patients and the public) and the environment whilist not limiting the possible beneficial human outcomes associated with ionized radiation exposure (1-3).

Each country is governed by regulations established by the highest competent authority, the International Radiological Protection System used throughout the world as a common basis for standards, legislation, guidelines, programs and practices of radiation protection. International Atomic Energy Agency (IAEA), which provides ideal recommendations in order to comply with the appropriate use of such radiation for peaceful purposes and beneficial to humans and the environment. In Paraguay, Law 2169/2014, also referred to as the Radiological and Nuclear Regulatory Authority (ARRN), has recently issued the Basic Regulation for Radiological Protection and Safety of Ionizing Radiation Sources, which specifies the requirements to achieve an adequate level of protection of people, property and the environment, against the harmful effects of exposure to ionizing radiation. Additionally, this law regulates the safety of involved sources, facilities and practices (4,5).

An installation where radioactive sources will be used requires the calculation of the necessary shielding to limit the doses received by workers and the general public (6). Paraguay, as a state member, has received the support of the IAEA and, on an institutional level, through the Nuclear Medicine Service Of the Department of Biomedical Engineering and Images of the Institute of Research in Health Sciences - National University of Asunción (IICS-UNA). The IICS-UNA has obtained support through technical cooperation projects, with the necessary contributions for the restructuring and implementation of the nuclear medicine service and the radio-pharmacy laboratory where the national and international requirements of building construction and radiological protection were contemplated for its habilitation.

The only available nuclear medicine service in the public health sector found at the IICS-UNA, uses SPECT (single photon emission tomography) technology which mainly relies on the radiopharmaceuticals marked as 99mTc. An expert mission was carried out with the collaboration of technical cooperation with the IAEA within the project **PAR 6014** "***Strengthening Nuclear Medicine for Diagnosis and Therapy***." The main objective of this mission included the advising on the design of a Nuclear Medicine Service that complies with the National and international radiation protection standards. This expert's mission culminated

in the drafting of a technical document which contemplates the calculation of shielding for all sectors involved in the management of ionizing radiation. Carried out in June 2013, this mission was a collaborative work between the experts of the IAEA along with the profesisonals and authorities of ICCS-UNA (7).

Currently the nuclear medicine service offers planar imaging obtained with the double-head gamma camera and also 3D reconstructions (SPECT). In addition, the cooperation project with the IAEA **PAR 6016 "*Providing Patients with Access to Public Nuclear Medicine Services for Early Diagnosis and Treatment*"** provides the coupling of a CT scanner and thus offers hybrid SPECT/CT technology for diagnosis by images, mainly benefiting patients of the public sector of the whole country.

OBJECTIVE

Disseminate the application of the shielding calculation adapted to the hybrid technology SPECT/CT, in the Service of Diagnosis by Nuclear Medicine of the Institute of Investigations in Health Sciences - National University of Asunción (IICS / UNA), with a view to ensuring the radiological protection of patients, the public and the environment.

METHODOLOGY

According to the report prepared by the IAEA expert, theoretical considerations have been elaborated for the estimation of the shields, which consisted in

1. Calculations
2. Design considerations.

The parameters used included dose rates (µSv / mAs), which were calculated by the methods of:

A) Use of up-and-down isodose curves supplied by the supplier (8) for SPECT / CT AnyScan SC. Using approximate values of voltage andcollimation.

B) Use of indexes known as CTDI (Indexed Dose Index) and DLP (Dose Length Product) tabulated by the National Council for Radiological Protection in its publication NCRP 147 (Table 5.2, NCRP 147) 9

Other parameters considered were: Estimated patient load per week, Transmission factors along with:

1. Design limits
2. Factors of occupation
3. Constructive data of walls and doors provided by those responsible for the buildingdesign
4. Factors of occupied areas surrounding the image acquisition room considering, in all cases, as radiations of significance, the radiation dispersed in the patient's body and the radiation leakage of the equipment.

RESULTS

The results of the evaluations yielded shield calculations made for each of the walls and doors, based on the data supplied by the manufacturer and the information provided by the institution. The information provided indicates that the CT shield is more than sufficient to perform the SPECT study.

DISCUSSION

Expert recommendations for verification suggest the use of real values that include the construction of an isodose curve and use of variables such as: Type/frequency of studies and type of radioisotope administered.

CONCLUSION

According to the theoretical works concerning the calculations that are recorded in detail in the final report written by the expert, it will not be necessary at this time to make any type of adaptation of the CT coupling.

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ASSESSMENT OF EXPOSURE AND SETTING OF LOCAL DIAGNOSTIC REFERENCE LEVELS FOR NUCLEAR MEDICINE PROCERURES AT VILNIUS UNIVERSITY HOSPITAL SANTAROS KLINIKOS

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Abstract

The assessment of patient doses in Nuclear Medicine (NM) is constant and mandatory process. According to international and national guidelines, recommendations it is important to periodically reassess local value of the diagnostic reference level (DRL). The purpose of this work was to assess prescribed measured activity of radiopharmaceutical for standard sized adult patients (70 ± 10 kg) and estimate effective doses. The study was performed at Vilnius University Hospital Santaros Klinikos during 2012 - 2017. The data of SPECT and PET/CT diagnostic procedures (N=274) was collected. For each procedure more than 20 patients were selected. The nuclear medicine procedures were performed with Philips Ingenuity TF PET/CT, GE INFINIA II, and GE INFINIA HAWKEYE4. For prescribed dose measurement different dose calibrators were used: Veenstra VDC-404, Veenstra VDC-405, COMECER ALTHEA-PC A together with B PITAGORA, COMECER IRIDE with PITAGORA. After patient exposure and dose estimation the correction actions were done and the local DRLs were established. Results showed that prescribed measured activities of radiopharmaceuticals were lower than national DRLs. In comparison with data from other healthcare institutions, the average doses of radiopharmaceuticals for Santaros Klinikos were slightly lower or at the same level.

1. INTRODUCTION

The practice of diagnostic nuclear medicine involves the use of ionizing radiation, and thus the potential risk associated with such exposure must be estimated, weighed against the benefits to the patient and optimised. According to international and national guidelines, IAEA recommendations, European Directive, countries should establish diagnostic reference levels (DRL) and it is important to assess patient doses and periodically reassess local values of the DRL in healthcare institutions, in order to manage optimal radiation dose for patients. Many investigations for a certain type of diagnostic nuclear medicine examination were done by different countries and hospitals. Those studies had shown a large spread of administered activities, even if similar equipment was used. In nuclear medicine, the DRL is given as administered activity for a standard size patient. The assessment of patient doses in Vilnius University Hospital Santaros Klinikos, (VUHSK) Nuclear medicine (NM) department is constant and mandatory process, which is an important tool for optimization.

The aim of this study was to perform the assessment of patient doses from NM procedures and establish the local DRL values for most common SPECT and PET/CT diagnostic procedures.

2. MATERIALS AND METHODS

The study was performed at Vilnius University Hospital Santaros Klinikos during 2012 - 2017. For each procedure more than 20 patients were selected. An effective dose for adults was evaluated by using conversion factor from ICRP publications multiplied by average administrated radiopharmaceuticals activities. For ^{99m}Tc-MIBI Myocardial perfusion scintigraphy scan (MP) one-day protocol were used 7.9E-03 mSv/MBq stress conversion factor, for the rest 9E-03 mSv/MBq (ICRP 53 and 80). For those procedures were used the conversion factors are from ICRP 53 and 80: ^{99m}Tc Thyroid scintigraphy 1.33E-02 mSv/MBq, for ^{99m}Tc-MAA Lung perfusion 1.1E-02 mSv/MBq, for ^{99m}Tc-MDP Bone scintigraphy 5.7E-03 mSv/MBq, for ^{99m}Tc-DTPA Dynamic renal scintigraphy 4.9E-03 mSv/MBq, for ^{99m}Tc-MAG3 Dynamic renal scintigraphy 7E-03 mSv/MBq. For ^{99m}Tc-MIBI parathyroid scintigraphy 8.5E-03 mSv/MBq (ICRP 62), ^{123}I-MIBG Neuroectodermal tumor scintigraphy 1.32E-02 mSv/MBq (ICRP 106), ^{18}F-FDG PET/CT tumor examination 1.9E-02 mSv/MBq (ICRP 106) [1-3]. For

estimating ED of ^{18}F-FDG PET/CT, computed tomography part was used conversion factor for the trunk (0.015 mSv/mGy/cm) [4].

NM procedures were performed with Philips Ingenuity TF PET/CT, GE INFINIA II, and GE INFINIA HWK4. For prescribed dose measurement following dose calibrators were used: Veenstra VDC-404, Veenstra VDC-405, COMECER ALTHEA-PC A together with B PITAGORA, COMECER IRIDE with PITAGORA dose calibrator. According to national and international standards dose calibrators are regularly checked [5]. They were meteorologically verified by Center for Physical Sciences and Technology with portable Capintec CRC-15R ionization chamber.

For ^{18}F-FDG examinations were used two systems: ALTHEA and IRIDE. ALTHEA is automatic fractionator with ion chamber for syringes in conjunction with wireless injection system. That system is a fully shielded hot cell with two dose calibrators. Automatic Infusion System IRIDE is also a fully shielded, motorized cart with the internal dose calibrator. The prescribed measured activity of radiopharmaceutical is syringed directly to the patient together with a standard volume of saline.

3. RESULTS

The patient exposure assessment studies were started at 2012 and after the analysis of the results the number of corrective actions, such us modified examination protocols, working procedures, calculations of prescribed activities and other optimizations were performed.

The nine common diagnostic scans constituting more than 86% of all NM examinations currently performed in hospital were chosen to propose local DRLs. This study included 274 SPECT and PET/CT diagnostic procedures. The collected data such as dose length product (DLP), the administered activity in MBq, patient data and effective doses were analysed.

Patient doses were estimated by ICRP publications 53, 62, 80 and 106 recommendations [1, 2, 3, 6]. Effective doses from SPECT and was estimated from the injected activity levels and corresponding conversion factors. The effective dose for whole body after PET/CT examination of CT component was assessed using the dose length product and were 361.6 ± 86.8 mGy*cm. Estimated average effective dose for PET/CT procedure was 5.42 ± 1.3 mSv. The results of VUHSK estimated effective doses, average effective doses of European countries [6], average administrated activities of VUHSK and the comparison with published data on the most common range of administrated activities from other Lithuanian healthcare institutions [7] and other European countries [8] are shown in Table 1. Local DRLs for most common procedures were calculated using the 75th percentile and comparison of local and national DRLs [9] for NM procedures are presented in Table 2.

TABLE 1. PATIENT ADMINISTRATED ACTIVITIES, EFFECTIVE DOSE AND ACTIVITIES IN OTHER EUROPEAN COUNTRIES

NM procedure	Most common value, MBq [8]	Lithuanian hospitals, MBq [7]	AAA* (MBq)	Effective dose, (mSv) [6]	Effective dose (mSv), VUHSK**
^{99m}Tc-MIBI MP 1 day (stress) (N=43)	1200 (for full exam.)	591 (for full exam.)	728.8±29.6	4.8	5.76
^{99m}Tc-MIBI MP 1 day (rest) (N=43)		-	363.2±13.5	5.5	3.27
^{99m}Tc Thyroid (N=33)	80	-	110.6±10.3	2	1.44
^{99m}Tc-MAA Lung perfusion (N=31)	150	517	100.4±6.7	1.8	1.10
^{99m}Tc-MDP Bone (N=31)	600	108	551.9±20.9	3.8	3.15
^{99m}Tc-DTPA Dynamic renal (N=22)	150-540	129	151.6±8.9	0.9	0.74
^{99m}Tc-MAG3 Dynamic renal (N=25)	100	367	92±3.6	0.8	0.64
^{99m}Tc-MIBI Parathyroid (N=30)	400-900	-	502.1±22	-	4.27
^{123}I-MIBG Neuroectodermal (N=21)	-	-	220,3 ±9,8	-	2.91
^{18}F-FDG PET/CT Tumor (N=38)	200-400	-	313.2±55.6	6.7	5.95

*- Average administrated activity of VUHSK

**- Effective dose estimated by ICRP 53, 62, 80 and 106

TABLE 2. COMPARISON OF LOCAL AND NATIONAL DRLs FOR NM PROCEDURES

NM procedure	Local (MBq)	National DRLs (MBq) [9]
^{99m}Tc-MIBI Myocardial perfusion	1111	1350
^{99m}Tc Thyroid scan	117	200
^{99m}Tc-MAA Lung perfusion	106	200
^{99m}Tc-MDP Bone scan	576	740
^{99m}Tc-DTPA Dynamic renal scan	160	370
^{99m}Tc-MAG3 Dynamic renal scan	95	100
^{99m}Tc-MIBI Parathyroid scan	515	740
^{123}I-MIBG Neuroectodermal tumor scan	229	400
^{18}F-FDG PET/CT Tumor scan	356	740

4. DISCUSSION

Our study results showed that for all types of analysed examinations, average administered activity was less than national DRLs. For some type of procedures such us ^{99m}Tc-DTPA Dynamic renal scan and ^{18}F-FDG PET/CT tumor examination administered activity was two times lower than DRLs. For ^{99m}Tc Thyroid, ^{99m}Tc-MAA Lung perfusion and ^{123}I-MIBG Neuroectodermal tumor scans administered activity compare to DRLs was lower from 53% to 58%. Diagnostic reference levels are indicators for the typical practice in a country or in a region, for this reason, radiation protection and health authorities in collaboration with professional medical organizations are responsible for setting diagnostic reference levels [10]. In the future, it will be wisely to reconsider the National DRL. A large variation between administrated activity ranges of Santaros klinikos, average administrated activity of 6 Lithuanian NM centers and other European countries can be seen in Table 1. The biggest differences were between ^{99m}Tc-MIBI Myocardial perfusion, ^{99m}Tc-DTPA Dynamic renal, ^{99m}Tc-MIBI Parathyroid scan and ^{18}F-FDG PET/CT Tumor examinations. For other administrated activities VUHSK level was closer to the lower administrated activity range edge. According to study done in the European countries average typical effective dose per diagnostic NM procedure (mSv) [6], compared with data from Table 1, shows that our results are comparable.

It is essential to evaluate patient dose, whereas it is an important parameter for optimization and monitoring of radiation exposure [1, 5, 10]. The estimated effective dose was inconclusive due to nonuniform distribution of activity within particular organs and using biokinetic data. Nevertheless, the effective dose is useful to estimate the risk of stochastic effects [2, 5].The effective dose gained from the CT component widely varies from 5 to 25 mSv and often exceeds the effective dose received from 18F-FDG injection[4].

5. CONCLUSION

The results of administered activities to patient in Lithuania were slightly lower than the data published in other studies, also prescribed measured activity of radiopharmaceuticals were significantly lower than national DRLs. According to physicians opinions, there is no doubt about examinations image quality. In VUHSK NM administered activities requiring for a good image during standard examinations are the main concern. The settled local DRLs based on administered activity is a tool for radiation protection and will play role for continuous improvement of clinical practice, NM patient exposure monitoring and dose optimisation ensuring appropriate image quality.

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IMPORTANT ROLE OF CZT ULTRAFAST CARDIAC CAMERA ON REDUCING RADIATION EXPOSURE IN MYOCARDIALPERFUSIONIMAGING

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Abstract

Aim: SPECT Myocardial Perfusion Imaging (MPI) is the most contributing nuclear medicine technique to the annual population dose. Novel camera technologies and reconstruction algorithms play important role on absorbed dose reduction. The aim of this study is to compare radiation absorbed doses to the patients examined by conventional SPECT (CSPECT) camera and ultrafast cardiac camera (UFCC).
Methods: MPI was performed with CSPECT (Optima GE) and with UFCC (GE NM 530c). Stress only/Two-Day stress/rest Tc-99m MIBI protocol was applied. For CSPECT camera imaging 740 MBq (20 mCi) and for UFCC imaging 185 MBq (5mCi) Tc-99m MIBI injected both for stress and rest imaging.
Results: Total effective dose to the patients were reduced from 11.69 mSv to 5.84 mSv when both stress and rest images were taken and from 5.84 mSv to 1.46 mSv when only stress images were taken by ultrafast cardiac camera .The mean total effective dose was significantly lower with UFCC (2.2±1.2 mSv) than CSPECT (7.7±3.8 mSv) ($p<0.001$).
Conclusion: This study showed that UFC cameras with the new detector technologies and the new reconstruction algorithms effectively reduced patients' radiation exposure from MPI.

1. INTRODUCTION

SPECT Myocardial Perfusion Imaging (MPI) is widely used for diagnosing coronary artery disease (CAD) and it is the most contributing nuclear medicine technique to the annual population radiation exposure. MPI needed to impart a lower radiation exposure to the patients, especially for the increasing number of patients referred for repeated ionizing radiation procedures and in the era of recent technical progress for dose reduction techniques in CT coronary angiography and increasing use of other non-ionizing modalities such as MRI and echocardiography. The aim of this study is to compare radiation absorbed doses to the patients examined by conventional SPECT (CSPECT) camera and ultrafast cardiac camera (UFCC) with Cadmium Zinc Telluride (CZT) detectors to investigate the role of novel camera technologies and reconstruction algorithms on absorbed dose reduction.

Increasing public annual radiation exposure has been a great concern during the last three decades. Medical radiation exposure (per capita per year) increased from 0.54 mSv in 1980 to 3.0 mSv in 2006 [1]. During this period number of nuclear cardiology studies increased significantly and contribution of nuclear cardiology studies to annual radiation exposure increased from 1% to 10.5% in the US (Fig. 1)

For reducing radiation exposure to the population appropriate use criteria was implemented to eliminate exposure in patients who do not qualify for nuclear imaging. American Society of Nuclear Cardiology (ASNC) created guidelines in 2010 to achieve the goal of doses of 9 mSv or below using Tc-99m agents for MPI. Patients' radiation exposure can be reduced by preferring Tc-99m agents to Tl-201, using stress first/stress only myocardial perfusion imaging (MPI) protocols, minimizing activity (mCi) to that needed to obtain good image quality and using improved reconstruction algorithms such as iterative reconstruction and resolution recovery.

2. METHODS

MPI was performed with CSPECT (GE Healthcare Optima) and with UFCC (GE Healthcare NM 530c). Stress only/Two-Day stress/rest and one-day Tc-99m stress-rest MIBI protocols were applied.

2.1. Radiopharmaceutical

A stress-first imaging sequence was employed using Tc-99 m sestamibi. If stress-first images demonstrated normal perfusion and normal left ventricular function, rest imaging was not performed. The two-day protocol was applied and 740MBq (20mCi) Tc-99 m MIBI injected both for stress and rest cardiac imaging when CSPECT camera was used. Two days stress-first imaging protocol was applied and 2.5MBq/kg (185-222MBq) Tc-99 m MIBI injected for stress and rest imaging when images were taken by UFC camera. If the one-day stress-rest protocol was applied 370MBq and 1110MBq Tc-99 m MIBI injected in CSPECT and 185MBq and 555MBq Tc-99 m MIBI injected in UFCC for stress and rest imaging.

3. RESULTS

Total effective dose to the patients was reduced from 12.4mSv to 3.1-3.7mSv when both stress and rest images were taken and from 5.8mSv to 1.4-1.7mSv when only stress images were taken by the ultrafast cardiac camera. The mean total effective dose was significantly lower with UFCC (2.5±1.2mSv) than CSPECT (9.1±3.8mSv) ($p<0.001$). Total effective dose reduced from 12.8mSv to 6.3mSv (almost %50 reduction) by using UFCC. (Fig.1)

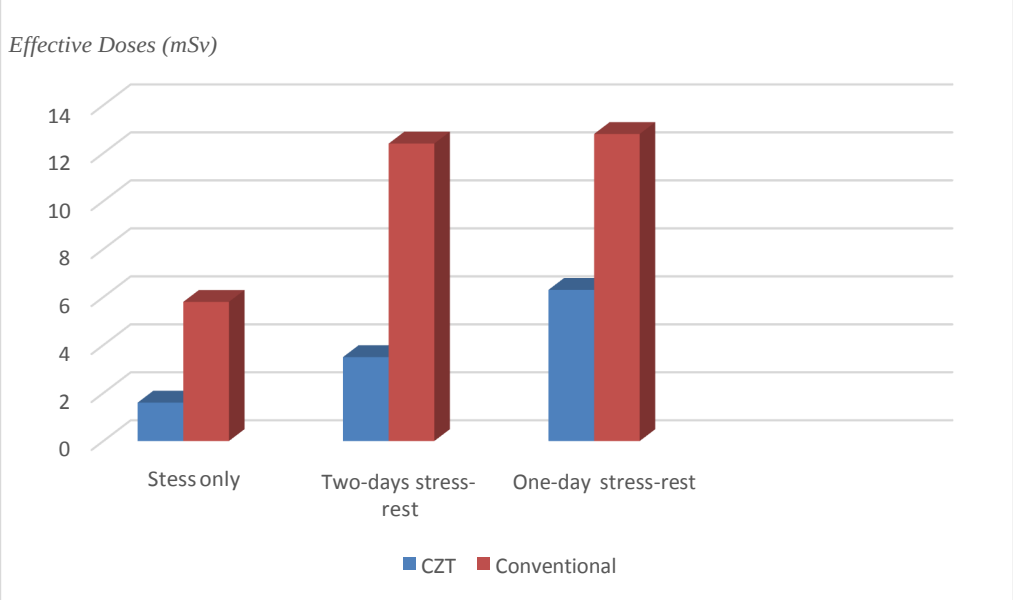

Fig.1. Comparison of effective doses (mSv) using UFCC and CSPECT camera for different protocols.

4. DISCUSSIONS

Several recent studies have conducted a dose reduction strategy with CZT cameras [2-8]. Although all of these studies applied similar but different imaging protocols, mean effective dose was less than 7mSv in all of them. In one study [7] mean effective dose was found 3.4mSv which is very similar to our study (3.1-3.7mSv).

In addition to dose reduction, UFCC image quality is better than CPECT cameras. This is due to better energy resolution and enhanced sensitivity of the specially designed collimators with CZT detectors when compared with CSPECT cameras with NaI-Tl crystal.

Iterative reconstruction algorithms which are used in UFCC also play an important role in dose reduction and improve image quality in low doses. The new systems use ordered subset expectation maximization (OSEM) iterative reconstruction algorithm which provides more accurate imaging models than filtered back-projection (FBP). This allows for higher resolution images with fewer counts and reduces injected activity and patient's radiation dose.

Using stress-only test the dose was reduced by 50 percent almost 30 percent of referred patients.

5. CONCLUSIONS

This study showed that when compared with conventional SPECT cameras, CZT UFC cameras with the new detector technologies and the new reconstruction algorithms effectively reduced patients' radiation exposure from MPI providing better image quality due to their much higher energy resolutions and count sensitivities.

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MEASUREMENT OF OCCUPATIONAL EXPOSURE DURING BONE SCINTIGRAPHY

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Abstract

Bone scintigraphy is used to evaluate the skeleton using radioactive substances. This study intended to evaluate radiation doses received by the nuclear medicine technologist bone scan procedure. Calibrated Thermoluminancent dosimeters (TLDs, GR200A) were placed at chest and hand of the three technologists during 21 bone scan procedures using MiE single head gamma camera after administration of 20 mCi of ^{99m}Tc-MDP. The mean chest and hand doses were 0.24 mGy, 0.63 mGy per procedure, respectively. It had been observed that, chest doses were higher because of direct handling with the injected patient. Increasing of staff awareness about radiation safety is recommended to enhance radiation protection and reduce unnecessary radiation dose. The staffs' organ surface doses are within the acceptable annual determined by the International commission on radiological protection (ICRP). Eye lens doses are slightly higher than the new established dose limits.

INTRODUCTION

Bone scintigraphy or bone scan is a diagnostic procedures used to evaluate the distribution of active bone formation in the body to detect cancer spread, infection, trauma or to assess the progress or deterioration after treatment. Bone scan procedure is one of the most regularly imaging procedure that carried out of all radionuclide imaging procedures. Bone imaging procedure using a radionuclide is widely available in many imaging centers, quick, relatively cheap diagnostic procedure that used in evaluation of plentiful pathologic conditions. There is an increase in the annual number of diagnostic nuclear medicine examinations worldwide, although the dose is not equally distributed worldwide (1 -3). It has been estimated the 24% of the global population living in countries with health care level I receive 90% of all nuclear medicine examinations (1). Sudan, which is classified as health care level III (1), has increasing number of nuclear medicine centers and many diagnostic procedures are performed using ^{99m}Tc. Bone scintigraphy was first introduced in the 1950s using calcium (^{45}Ca) and strontium (^{85}Sr) (3). The introduction of the ^{99}Mo/^{99m}Tc generator and gamma camera in 1950s was enabled clinician to perform bone imaging procedures (3). In diagnostic nuclear medicine procedures, the main contribution to the collective dose arises from ^{99m}Tc bone scan, ^{201}TL cardiovascular studies and Iodine thyroid scans (1-3). The main agent currently used for bone scanning is ^{99m}Tc methylene diphosphonate (MDP); following intravenous injection MDP circulates in the vascular system for a two to three hours before whole body imaging procedure (4).

Nuclear medicine involves handling of radioactive materials that can give rise to external and internal exposure of staff to the whole body consequential from eluting and preparing the radiopharmaceutical materials in hot lab and administration the ^{99m}Tc/^{99}Mo to, as well as from, patients after the procedure which imposes radiation risks for both patients and staff (1,2). As a consequence, unprotected parts by the lead apron, may receive to un avoidable radiation doses from different radiation source depending on workload and procedural technique. It has been reported that staff may receive 18.6 mSv annually (3). The magnitude of exposure depends on radionuclide, its activity and type of work within a department in which the person is involved (1). The international commission on radiological protection (ICRP) (5) has lowered the annual dose limit for the eye lens from 150 mSv to 20 mSv (i.e., by a factor of 7.5) for occupational exposures, therefore it is crucial to evaluate it s impact in the existing programs on radiation protection and safety.

Therefore, radiation protection in nuclear medicine is of special concern due to use of unsealed sources of radiation. In addition to that, still limited data available worldwide (2,4,6,7,8).The previous studies showed large dose variability due t o difference in imaging protocol, imaging techniques and operator skills. To our knowledge, and this is the first study intended to evaluate staff exposure during bone scan in Sudan. This study is aimed to evaluate the radiation doses received by the nuclear medicine staff during bone scan procedure.

MATERIALS AND METHODS

Occupational exposure was measured for three technologists during 21 bone scan procedures**.** The clinical indications of the procedures were bone metastasis due to prostate breast cancer. The ethics and research committee approved the study and informed consent was obtained from staff and patients prior to the examination .A total of 50 thermoluminescence dosimeter (TLD-GR200A) circular chips of lithium fluoride (LiF:Mg,Cu,P) were used in this study. Prior to measurements, all TLD were calibrated in terms of air kerma free-in-air under reproducible reference condition (9). Background signal was subtracted prior dose calculation. The TLD signal was read using an automatic TLD reader (Fimel PCL3, France) in an atmosphere of inert nitrogen. The stability of the reader was checked before any reading session. The read-out was at a 155 ^{0}C preheat temperature and the signal was acquired from 155 to 260 ^{0}C with heating rate of 11^{0}C/s. Three TLDs were placed in a plastic bag made of transparent polyethylene plastic foil placed on organ site and were fixed in the required position with adhesive tape. Staff effective dose and eye lens dose was extrapolated for chest surface dose using the methodology and tissue weighting factors reported in ICRP 103 according NCRP 122 report (10, 11, 12). Staff height was recorded since it affect the amount of scatter radiation reaches their eyes due the inverse square law. Before each irradiation all dosimeters were annealed in an annealing oven (TLDO; PTW, Freiburg, Germany) at 240°C for 10 min.
Radiation doses were measured at chest and right hand of three technologists during 21 bone scan procedures. The technologists performed the elution of ^{99m}Tc and imaging process. The technologists wear a lead apron during all procedures. No goggles of thyroid collar were used. All procedures were performed at Al-Nilain diagnostic center, Khartoum, Sudan. The mean administered radiopharmaceutical was 20.0 mCi of ^{99m}Tc-MDP based on patient weight. The eluted $NaTcO_4$ is mixed with methylene diaphosphonate MDP and administrated intravenously to patient before 2 to 3 hours before scanning. Patient scanned using Single head gamma camera (Medical imaging electronics MiE). Quality control performed before administration of the radiopharmaceutical and doses are carefully calculated. The data acquisition is obtained on supine position for 15 minute and prone position for 15 minute for anterior and posterior views of the axial skeleton, with scanner speed 17 cm/min , the adjusted length is 190 cm, the field 408 × 408 and the matrix is 256 × 256.

RESULTS

Bone scan procedure, which is used for used to evaluate the distribution of active bone formation in the body, is the most common imaging procedures in our nuclear medicine department. The bone scan procedures were performed for 21 patients (11 (52.3%) males, 10 (47.7%) females for different clinical indication. The main clinical indications for this procedure were prostate and breast cancers (47.7%) while the other clinical indications include thyroid, colorectal, lung and stomach cancers. The patients' age was ranges between 31.0 – 71.0 years and patients weight 48.0 to 105.0 kg. The projection image of the skeleton illustrates the distribution of radionuclide uptake (metabolism) (Figure 1).

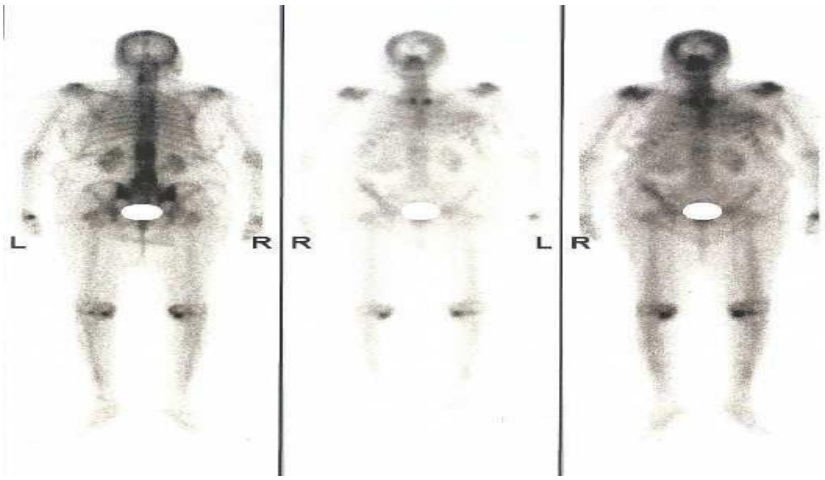

Figure 1: bone scans of metastatic breast cancer for 60 years old female using 18.2 mCi

It is well known that tumors has the ability to accumulate Tc ^{99m}.Thus evaluation of the organ doses, for the nuclear medicine staff during whole body bone scan procedure is vital in order to evaluate the protection and radiation risks

Table 1. Staff dose (mGy) during bone scan

Organ	Minimum	Mean ± Sd	Median	Maximum
Hand	0.14	0.24±0.34	0.39	0.82
Chest	0.30	0.63±0.33	0.54	1.71

Table 1 shows the mean hand and chest doses for the three nuclear medicine technologist were 0.24±0.34 mGy, 0.63 ±0.33 mGy per procedure, respectively. The mean radiation dose to the chest doses was higher than the mean hand dose during bone scan procedures. The mean height of staff 164(155-188) cm. Staff exposure is dependent to the distance from the patients.). Figure 2 show that the staff organ doses, which ranged between 2.5 µSv to 25.0 µSv per procedures. The variation of organ equivalent doses is depending on the distance from the patient and the depth of the organ.

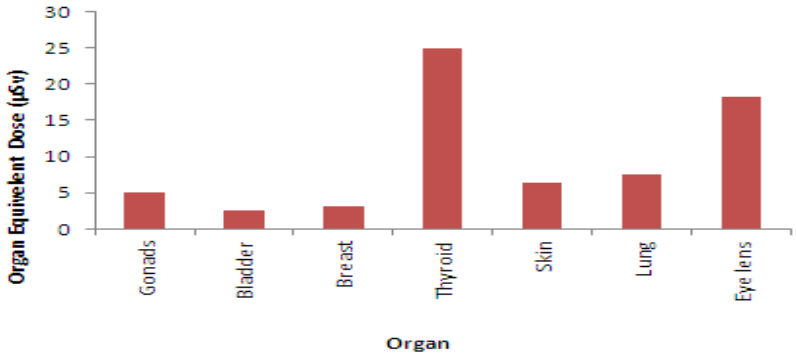

Figure 2. Organ equivalent dose during bone scan procedure

DISCUSSION

Nuclear medicine technologists are exposed to radiation to various locations during various stages of image acquisition (5). Therefore, staff received a significant radiation dose. This study revealed that the the mean hand and chest doses for nuclear medicine technologist were 0.24±0.34 mGy, 0.63 ±0.33 mGy per procedure. This can be attributed to the injected patient's spent long time with technologist before and during scanning procedure. It is important to note that skeletal phase images are typically acquired from 2–5 h after radionuclide administration. Delayed scans up to 24 hours could be obtained, based on the clinical indications (13). The same result has been reported previously that the external ionizing radiation exposure of nuclear medicine technologists arises primarily from radioactive patients rather than from the preparation and injection of radiopharmaceuticals (13,14). The average dose for a scan depended not only on the administered activity and isotope but also on the amount of patient contact required. According to the patient flow of the department, a total of 1248 bone scan procedures per year. Consequently, the hand dose will be 300.0 mGy procedures which is the dose limit of 500 mSv which determined by the International Commission on Radiological Protection (10).

Figure 2 shows that the staff organ doses ranged between 2.5 µSv to 25.0 µSv per procedures. The mean eye lens dose was 18.2 µSv per procedures. Therefore according to the current workload, the annual dose to the eye lens was estimated to be 22.7 mSv. The annual eye lens doses are slightly higher than the new eye lens dose limit. However, if the technologist worn a lead glass, the eye lens dose will decrease 90% of its original value (15). Thyroid dose is relatively high because no one of staff is using thyroid shield. It has been reported that 0.5 mm lead equivalent shield with attenuate enough of the medium-energy photons emitted by Tc^{99m} (140 keV) to 70% (15). These results indicated that the staff is well protected by lead apron. In the literature, staff doses were measured during specific period of time for different types of procedure and different radiation energies. The overall annual effective dose for technologist (15.4 mSv) is within the dose limit determined by ICRP (20 mSv). The previous studies suggested that exposure was not likely to exceed the annual limit for extremities and eye lens according to their workload (2,4,6,7,13,16). Piwowarska-Bilska et al (2), reported that technicians are the largest occupational group in nuclear medicine department (18.6 mSv) because of the fact that they prepared radiopharmaceuticals,

performed examinations of the patients and controlled the scanners. The International Atomic Energy Agency (IAEA) recommends that the average annual dose for exposed workers in a nuclear medicine facility should range from 3.0 to 5.0 mSv (17). From the previous results, it is very clear that technologist occupational dose depend on the workload and protection shields (goggles, thyroid shield and lead apron). The use of additional radiation protection measures such as revision of the department layout, use of mobile shields, revision of scan protocol and proper personnel protection equipment are essential during nuclear medicine scans.

CONCLUSIONS

Staff eye lens is exposed to radiation dose 12% higher than the dose limits. Organ doses under the lead apron have demonstrated that the doses are well below the recommended levels of annual occupational radiation exposure. There is a wide variation among the radiation dose received at hands, eyes and thyroid glands which may be attributed to the difference in patient metabolism, age and weight. The use of thyroid lead shield and lead glass with reduce the dose. Staff training, the use of shields or increasing the distance, reduces contact time with the patients after injection and workload reduction are effective methods for dose reduction. The results of this study showed that technologist protection in nuclear medicine department must be revaluated in the light of ICRP eye lens dose limit recommendations.

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INTERNAL DOSIMETRY ASSESSMENT IN INDONESIA: REPLYING THE BONN CALL FOR ACTION FOR ORGAN DOSE ASSESSMENT

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Abstract

One of action in Bonn Call for Action has stated that the methods in organ dose assessment in radionuclide therapy should be improved, as well as in external radiotherapy. Since the organ dose assesment has not been conducted yet in Indonesia, to answer the action, a series of promotions and works related to the internal dosimetry assessment have been done in the Center for Technology of Radiation Safety and Metrology since 2013. In addition, the research in new radiopharmaceutical in Indonesia has increased in number, and need to be attributed by internal dosimetry studies to estimate the organ dose due the administration of unsealed radioactive source into human body. This paper has been addressed to evaluate the progress and consider the future works in organ dose assesment in Indonesia. As a result some works have been done, including the preparation of dosimetry protocol, establishing anthropomorphic phantoms to develop dosimetry protocol, predicting the organ dose in human based on preclinical studies, and evaluating the recommendation of administered dose for patients in Indonesia. However, all the works has been done in the centre should be expanded to the hospitals so the impact will be directly addresed to patients.

1. INTRODUCTION

In Indonesia, radiation therapy using unsealed source has been done in several hospitals with Nuclear Medicine services, but the assessment for organ doses has not been done previously. As it has been stated in the Bonn Call for Action in 2012, the organ dose assessment for unsealed radioactive source is one of safety aspect in radiation therapy which should be improved as well as has been done in external radiation therapy [1]. To answer this action, there was a series of promotion and works regarding internal dosimetry assessment in Nuclear Medicine in Indonesia that has been started in 2013[2]. This article is addressed to evaluate the progress and consider the future works in internal dosimetry assessment in Indonesia

The works in internal dosimetry assessment basically has been referred to the Act No. 10 in 1997 regarding nuclear energy. The law has stated that in any nuclear energy application in Indonesia should consider the benefit which must be greater than the risk [3]. Therefore, in the case of administration of unsealed radioactive source in human body, internal dosimetry will be useful as a tool to assess the risk. Moreover, the risk will increase in therapeutic nuclear medicine procedure in which a greater dose will be administered to treat the diseases, such as cancer and other diseases which is preferably to be treated with nuclear medicine procedures[4].

For injection of radioisotopes with the goal of therapy (Radioisotope / Radionuclide Therapy), internal dosimetry assessment might be used as a treatment planning [5] as well as in external radiotherapy. With the same purposes, treatment planning in radionuclide therapy will assess the organs at risk (OAR), and spare health tissues. For example, the administration of 177Lu in Peptide Receptor Radioisotope Therapy (PRRT) has considered kidneys as the organ at risk and set up the kidney as dose limiting organ, which means that the administered dose could not cause the kidneys receiving the organ dose more than the limiting dose for kidneys. The consideration would be taken to prevent the kidney from radiation nephropathy, which might result the kidney damage due to high radiation [6]. Nevertheless, since the internal dosimetry in Indonesia only has been promoted in 2013, the

works related internal dosimetry assessment has been initiated in the same year and the response from the Nuclear Medicine clinicians is favourable.

2. IMAGE QUANTIFICATION FOR DOSIMETRY PROTOCOL

To perform internal dosimetry studies in Nuclear Medicine procedures, it is impossible to measure the organ dose directly, so performing image acquisition and being quantified further are the approaching method to investigate how the uptake of radiopharmaceutical has been distributed into human body[7]. One of well known method in image quantification has been pronounced by Medical Internal Radiation Dosimetry (MIRD) committee by publishing the MIRD Publication no. 16 about the techniques for quantitative radiopharmaceutical biodistribution data acquisition and analysis for use in human radiation dose estimation [8]. This method has utilized a pair of anterior and posterior whole body scan from gamma camera planar. Additional features such as, background correction, attenuation factor, and the calibration factor of gamma camera also were needed. The analysis of image quantification will result an organ activity, as follow:

$$A_j = \sqrt{\frac{I_A I_P \, f_j}{e^{-\mu_e t} \, C}} \qquad (1)$$

With :

A_j = organ j activity (MBq or mCi)
I_A, I_P = count rate of anterior image and posterior (cps)
= transmission factor of patient with t thickness
C = calibration factor (cps/MBq) or (cps/mCi)
f_j ≤ 1 , correction factor of t and attenuation factor
μ = attenuation factor (cps^{-1})

A series of image quantification studies has been done in few publications [9,10] to initiate the internal dosimetry assessment in Indonesia. The works also have been presented in such seminar to continuously promote the organ dose assessment in Nuclear Medicine procedures. Moreover, a procedure of image quantification in the centre has been established to emphasize how important the organ dose assessment as a tool for radiation protection in patients.

3. ESTABLISHING ANTHROPOMORPHIC PHANTOMS

Anthropomorphic phantom in Nuclear Medicine is needed for investigating the characteristic of interaction between the radiation in human body and the detectors of Gamma Camera. The forms of phantoms may vary started from simple form to the similar shape and size of organs [11].

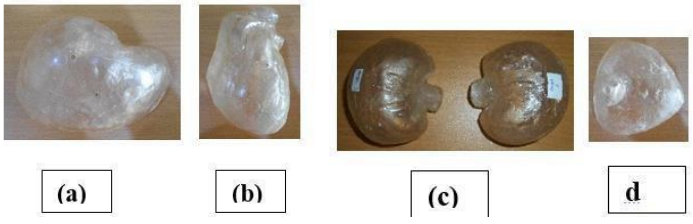

Fig.1 Anthropomorphic phantom (a) liver, (b) heart, kidneys(c), bladder (d)

Regarding the need of internal dosimetry studies, anthropomorphic phantoms has been produced based on the anatomical data from the reference [12]. The phantoms are made from the PMAA (resin) with the coefficient of linear attenuation close to water and regarded as tissue equivalent materials. The investigation has been done

by filling the phantoms with ^{99m}Tc and perform the acquisition of phantoms for various amount of activities. The evaluation and accuracy analysis have been done by using eq, (1)

The result of accuracy evaluation have shown that the difference between the activity as a result of image quantification and the activity from dose calibrator is less than 20% and still better than previous study [13]. Another study also has suggested the CT based correction should be implemented to reduce the difference value in image quantification [14]. Hence, SPECT/CT and PET/CT would be better to be utilized for establishing anthropomorphic phantoms as an additional tool for organ dose assessment in nuclear medicine procedures.

4. PREDICTING THE HUMAN DOSE IN NEW RADIOPHARMACEUTICALS

Since there is increasing number of new radiopharmaceutical production, internal dosimetry assessment in development of new radiopharmaceuticals also has been considered as raising issue in Indonesia. Few studies [15-18] have reported that the data from biodistribution tests in the animals might be useful for estimating the distribution of radiopharmaceutical in human body by using the scaling conversion from the animal into human by following the scaling factor conversion [18]:

$$\left(\frac{\%ID}{org}\right)_H = \left[\left(\frac{\%ID}{org}\right)_A x(kg_{TB})_A\right] x \left(\frac{g\ org}{kg_{TB}}\right)_H \quad (2)$$

The conversion will give the estimation of injected dose in human, then it can be used to estimate the residence time of the radiopharmaceutical and will be useful to predict the human dose as effective dose. In the case of diagnostic radiopharmaceutical, the effective dose might be useful for predicting the risk, but in the case of therapy, the effective dose might be useful to find the dose limiting organ to define the number of injected dose should be delivered for the treatment and applied as a treatment planning [19].

5. EVALUATING THE RECOMMENDATION OF ADMINISTERED DOSE

It has been reported that the variation of organ size within models in Nuclear Medicine reference group has affected the effective dose which will be received [20]. For example, the variation of organ size between Asian and Caucasian will produce different value of effective dose [21]. For this reason, a review about the administration dose of Tc-99m in various diagnostic procedures in Indonesia has been done by referring the organ size of Caucasian and Asian model [22]. Since Indonesia has no standard model, a Tanaka model has been referred and reviewed to present how much the variation of organ size will affect the effective dose. In addition, the Nuclear Energy Regulatory Agency (BAPETEN) Indonesia has adopted the medical guidance for nuclear medicine procedures from IAEA Safety Guide [23] into the Decree of BAPETEN Head No.17 year 2012 regarding the radiation safety in nuclear medicine service. The value in medical guidance has been referred from ICRP standard or Caucasian model, so when the standard has been applied for Indonesia/Asian group, the procedures might give higher effective dose to the patients in Indonesia, which means it will increase the stochastic risk. Hence, it would be better if the maximum administered dose in the Decree of BAPETEN Head No.17 year 2012 need to be reviewed to reduce the probability of stochastic risk for the patients in Nuclear Medicine Department in Indonesia.

6. CONCLUSION

It is clear that a series of activities which has been done in internal dosimetry assessment in Indonesia, can be reflected as the answer of the action no.5(f) in Bonn Call for Action. Despite the promotion, research and works have been applied only in the Centre for Technology and Radiation Safety Metrology, but some published papers have shown that the works still on going and will be continued to implementing institutions such as hospitals and/or academic institutions in the near future. It is also expected that the Regulatory Agency will be more aware that the organ dose assessment is a global raising issue for better pateint safety in nuclear medicine services. In addition, there are also a series activities such as additional lectures, seminars and workshops which have been performed outside the centre to introduce, promote and find possible collaborated projects, so that the organ dose assessment in Nuclear Medicine procedures will be developed more as well as the organ dose assessment in External Radiotherapy procedurs.

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PhD Project Research

PET/CT imaging with a [68Ga] Gallium-labelled PSMA Ligand for the Diagnosis of Prostate Cancer in Nuclear Medicine Department & Radiation Protection

Abstract

Prostate-specific membrane antigen (PSMA) is a cell surface protein with high expression in prostate carcinoma (PC) cells. Recently, procedures have been developed to label PSMA ligands with 68Ga. Our initial experience (68Ga-PSMA) suggests that this novel tracer can detect PC relapses and metastases with high contrast.

The aim of this study is to show that 68Ga-PSMA presented with excellent contrast as early as 40 min and 1 hr post injection with high detection rates even at low PSMA levels compared to [^{18}F] Choline, investigate the 68Ga-PSMA biodistribution in normal tissues and tumor tissues, and the results will be compared between the involved hospital (AUBMC) and the International Reference Levels of patient dose. This study will help provide a more unified diagnostic radiology practice and help to reduce patient exposure levels to those comparable to other international standards.

Method: A total of 70 patients with PC and rising prostate specific antigen (PSA) levels were subjected to 68Ga-PSMA and [^{18}F] Choline positron emission tomography (PET)/CT.

- Measure the dose after the injection at the surface and at 1(m).
- Measure the dose before the discharge of the patient at surface and at 1(m).
- Analyze of mean and maximum standardized uptake values (SUV mean/max) of the Kidneys and the bladder.
- Steps of protection of patient, workers, people and environment according to the ALARA principle.

Results:

- 68Ga-PSMA for prostate cancer presents excellent contrast and a high detection rate even when the level of prostate specific antigen is low compared to those of [^{18}F] Choline.
- Quantitative assessment revealed excellent contrast between tumour lesions and normal tissues.
- Absorbed dose taken by the patients due to Ga-68 is much lower than that taken by [^{18}F] Choline
- Using 68Ga-PSMA in the diagnostic of prostate cancer instead of [^{18}F] Choline.

^{68}Ga-PSMA ligand PET/CT in patients with prostate cancer:

Background: 0.018 (μ Sv)

Median age: 70 (53-83)

Patient	Mass(kg)	Dose (mCi)	Dose at injection (μ Sv)/after 15 min		Dose before discharge (μ Sv) (after 1 hr)	
			surface	1(m)	surface	1(m)
1	68	3.42	Dynamic: 82	8	28	7
2	77	2.83	Dynamic:51	14	19	12
3	74	2.93	Dynamic:51	14	20	11
4	77	3.57	40	11	23	9
5	89	27.1	Dynamic:74.2	11.4	34	12.3
6	75	3.61	88.2	11.2	36.4	11.2
7	71	3.35	Dynamic: 79	11.3	24	11
8	75	3.62	Dynamic:53	14	21	12
9	82	2.98	Dynamic:77	13.5	19	11
10	67	3.34	Dynamic:57	14	22	11.4
11	88	2.81	83	11	22	9
12	77	2.93	82	11.2	23	9
13	82	3.62	76	11	19	10.1
14	81	3.55	Dynamic:57	11.4	19	11
15	76	3.61	81	14	32	11
16	74	2.98	Dynamic:78	14	25	11
17	80	3.45	Dynamic:75	14	22	12.5
18	66	3.40	Dynamic:82	11	20	9
19	73	3.61	Dynamic:79	12.9	19	10.3
20	75	2.91	Dynamic:74	11	19	9
21	81	2.84	69	11	20	9.4
22	69	3.34	86	11	32	10.1
23	76	3.45	Dynamic:73	14	19	12.1
24	83	3.31	85	14	23	9
25	77	3.36	88	14	23	9
26	75	3.63	83	14	20	12.1
27	84	2.97	Dynamic:79	11	19	9
28	86	3.35	Dynamic:75	14	21	12
29	75	3.32	Dynamic:69	14	19	12.5
30	80	2.92	Dynamic:74	13.1	23	9
31	81	2.91	Dynamic:77	14	21	12
32	85	3.63	87	14		12
33	74	3.31	86.4	11		9
34	71	3.32	88.3	14		11

Patient	Mass(kg)	Dose (mCi)	Dose at injection (μ Sv)/after 15 min		Dose before discharge (μ Sv) (after 1 hr)	
			surface	1(m)	surface	1(m)
44	67	3.51	Dynamic:53	13.1	19	9
45	87	3.32	Dynamic:81	14	37	11.9
64	82	3.34	Dynamic:78	14	19	12.3
74	71	3.35	Dynamic:75	14	19	11
84	73	3.63	Dynamic:55	11	19	9
94	80	3.61	Dynamic:78	11	23	9
50	77	2.98	87	14	36	11
51	71	3.34	Dynamic:52	14	19	11
52	75	3.54	Dynamic:51	11.3	19	10.6
53	84	3.12	88	11	21	9
54	86	2.95	76	11.4	19	9
55	70	3.34	Dynamic:79	14	20	12
56	71	3.62	Dynamic:71	14	19	12
57	68	3.55	Dynamic:73	14	19	11.1
58	79	3.32	86	11	28	9
59	74	2.97	Dynamic:54	11	19	9
60	86	3.31	Dynamic:51	11.2	19	9
61	73	3.37	87	11	23	9
62	67	2.87	83.6	14	35	12
63	79	3.36	88	13.4	31	10.9
64	87	2.98	Dynamic:78	14	19	12.1
65	81	2.87	Dynamic:72	14	19	12
66	79	3.34	Dynamic:79	14	21	12.9
67	76	3.52	Dynamic:51	11	20	9
68	65	3.61	84	11	23	10.1
69	85	3.33	68	14	19	12
70	77	3.31	87	11	21	9

FDG- PET/CT in patients with prostate cancer:

Measuring the dose after 30 minutes of injection:

Background: 0.018 (μ Sv)

Patient	Mass(kg)	Dose (mCi)	Dose at injection (μ Sv)/after 15 min		Dose before discharge (μ Sv) (after 1 hr)	
			surface	1(m)	surface	1(m)
1	82	8.15	240	15	101	15
2	86	8.53	250	20	111	12
3	75	5.71	230	14	94	13
4	56	5.74	235	14	103	12
5	77	8.41	250	14	92	12
6	71	8.35	250	14	98	12
7	75	8.41	234	14	110	13
8	84	8.71	230	22	101	15
9	86	6.97	229	14	115	12
10	70	5.74	241	15	92	12
11	71	5.97	250	21	107	13
12	68	6.69	239	14	111	13
13	79	8.18	230	14	104	12
14	74	8.35	242	14	117	12
15	77	8.24	247	15	98	11
16	75	6.98	230	20	97	15
17	84	7.37	251	14	105	9
18	86	8.47	250	14	110	12
19	67	8.12	235	15	112	12
20	88	8.36	239	14	110	13
21	77	6.78	230	14	98	12
22	82	8.34	235	15	97	12
23	68	7.96	233	20	98	15
24	77	8.19	245	14	103	12
25	74	7.98	250	14	101	12
26	77	6.23	245	14	110	13
27	89	5.78	230	15	98	12
28	75	5.57	250	15	94	15
29	71	8.34	250	20	96	15
30	75	8.31	240	14	110	12
31	82	8.31	245	15	111	13
32	73	7.65	250	14	109	12
33	84	5.96	230	14	98	12
34	85	5.82	240	20	92	13

Patient	Mass(kg)	Dose (mCi)	Dose at injection (µ Sv)/after 15 min		Dose before discharge (µ Sv) (after 1 hr)	
			surface	1(m)	surface	1(m)
44	87	8.31	250	14	101	12
45	82	8.22	250	14	107	12
64	71	5.81	250	23	98	15
74	73	7.83	234	14	94	13
84	80	7.91	245	14	104	12
94	69	8.29	248	15	110	12
50	76	7.74	246	14	104	12
51	83	5.57	230	20	111	11
52	77	8.13	251	23	98	12
53	75	7.89	249	14	110	12
54	84	7.41	230	14	98	12
55	86	6.34	250	15	98	12
56	75	6.85	250	14	103	11
57	80	8.81	240	15	97	12
58	76	8.49	234	20	102	12
59	74	8.32	231	14	105	12
60	80	7.75	245	15	11195	13
61	66	7.61	230	14	98	13
62	73	8.15	240	21	106	15
63	75	8.18	240	14	111	12
64	81	8.26	230	14	95	12
65	69	5.73	250	20	99	13
66	76	6.97	250	14	110	12
67	66	8.31	230	14	107	12
68	83	8.26	234	15	98	12
69	74	8.28	250	14	113	12
70	87	6.74	250	14	94	12

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COMPARISON OF STAFF'S RADIATION EXPOSURE – LIQUID VS. CAPSULE 131I

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Abstract

Treatment of thyroid cancer with radioiodine (131I) is a trusted medical method for decades. As such, it carries the risk of external and internal irradiation to which the staff of the concerned department is exposed. The risk is the direct contact with the radiopharmaceuticals ($Na^{131}I$), with the patients after application, or with the contaminated waste. Lab workers, in whose lab the radiopharmaceutical is prepared for administration to individual patients and the residual amount of radioiodine is processed, are especially exposed to direct contact with radioiodine.

Comparison of workers' radiation exposure from liquid and from capsule was based on evaluation of personal and finger dosimeters and measurements of internal thyroid contamination. The monitoring period was from September 2012 to the end of August 2016. Data from personal dosimeters were also evaluated in relation to the amount of activity applied during these periods.

Four radiological assistants and one technician who work at the Department of Nuclear Medicine and Endocrinology at Motol University Hospital (DNME) were selected to assess the radiation exposure. In the case of chosen workers, the decrease of contamination of thyroid and of dose equivalent in the order of dozens of percent occurred after the transition to capsules. Such difference in irradiation was not observed in other workers of the DNME.

1. INTRODUCTION

The aim of this work is to compare the radiation exposure of the staff most likely to be in contact with ^{131}I within four years (from September 2012 to the end of August 2016). In the first two years, the ^{131}I liquid was administered to the patients at Department of Nuclear Medicine and Endocrinology at University Hospital Motol (DNME). In the following two years, patients were given ^{131}I capsules. The influence of the change of the pharmaceutical form of radiopharmaceuticals as well as the number of administered radiopharmaceuticals on the irradiation was monitored by the attending staff.

Usage different forms of radiopharmaceuticals can lead to an increase or decrease in the likelihood of contamination of workplace surfaces, protective equipment and workers. Radiopharmaceutical-related activities have been significantly simplified following the usage of ^{131}I capsules, and the average time of exposure of radiation workers was significantly shortened. [1–2]

2. METHODS

For radiation workers' exposure assessment, four annual time segments were selected, from September to August of the following year (starting in September 2012). In August 2014, there was an irreparable damage to the pipette which served to process the activity of the ^{131}I liquid in the laminar box occurred. As of September 2014, another drug form of radioiodine (capsule) was then administered.

For the given periods, the results of the thyroid contamination measurements were monitored by a scintillation probe. A probe calibration coefficient was determined and yearly checked for radioiodine activity in the thyroid determination. Monitoring with this measuring kit was performed routinely after 14 days for each worker who treated ^{131}I. In addition, the monthly personal dosimetry results (whole-body dosimeters operating on the principle of optically stimulated luminescence OSL and finger thermoluminescence dosimeters TLD)

were compared. The dependence of the personal dosimetry results on the amount of activity administered, the number of applications and the route of administration during these periods was discussed. The amount of administered activity submitted over the whole period was tracked based on application protocols.

Initial monitoring was performed by all employees of the ward. After thorough study of the data, it was found that a significant change in the radiation exposure occurred only for workers of the radioiodine laboratory. Radiology assistant pipetting activity, preparing solution for individual patients, and then measuring retention and performing scintigraphy examination. The technician participated in the liquid radiopharmaceutical maintenance and piston for dosing the radioiodine in the pipette check. The transition to capsules with fixed activity limited activities in direct contact with the radiopharmaceutical. In total, 5 DNME employees were evaluated in detail - 4 radiological assistants and 1 technician. Only two employees (1 radiological assistant and 1 radiology technician) worked at the workplace during of wholeradiation monitoring.

3. RESULTS

3.1. Amount of activity

During the four-year study period, a total of 11.47 TBq of radioiodine was administered to patients at DNME. 10.60 TBq, out of the total amount was used for therapeutic applications, 0.31 TBq for diagnostic applications and 0.56 TBq for ^{131}I-mIBG therapy. The amount of activity ^{131}I given in each period is shown in Table 1. It is clear from the table that there has been an increase in the number of therapies administered after the introduction of the capsules. However, this is not related to the change in the dosage form but to the change of treatment strategy at DNME.

TABLE 1. COMPARISON OF AMOUNT OF ADMINISTERED ACTIVITY OF ^{131}I IN REPORTING PERIOD.

	Amount of applied activity [GBq]			
	9/2012 – 8/2013	9/2013 – 8/2014	9/2014 – 8/2015	9/2015 – 8/2016
Diagnostic	87	77	76	59
Therapy	2503	2399	2851	2846
^{131}I-mIBG	141	147	202	70

Of the total number of applications, there is a clear trend towards reducing the number of diagnostic applications compared to the number of therapeutic applications (which is related to the optimization of the number of diagnostic control applications one year after the treatment). The total number of administration remains the same when switching from liquid to capsules, averaging 1260 per year. The exact number of ^{131}I applications over the last four years is recorded in Table 2.

TABLE 2. NUMBER OF DIAGNOSTIC, THERAPEUTIC AND ^{131}I-MIBG ADMINISTRATIONS 131I OVER FOUR CONSECUTIVE YEARS.

	Number of applications ^{131}I			
	Liquid		Capsule	
	9/2012 – 8/2013	9/2013 – 8/2014	9/2014 – 8/2015	9/2015 – 8/2016
Diagnostic	687	697	599	537
Therapy	573	559	711	674
^{131}I-mIBG	32	34	31	14
Total	**1292**	**1290**	**1341**	**1225**

3.2. Results of thyroid contamination measurements and personal dosimetry

Comparing the results of the thyroid contamination monitoring revealed the fact that the use of the capsules reduced the radiation exposure of selected workers. For workers who come in direct contact with pharmaceuticals, the measured thyroid contamination dropped by several tens of percent. The transition from one drug form to another did not lead to a change in thyroid contamination in other DNME workers. The highest decrease of radioiodine activity in thyroid was observed for the technician. The main reason for the decline is that the use of capsules significantly reduced direct contact with radiopharmaceuticals (repairs and maintenance of the shielded box and pipette). The decrease in contamination was over 90 %. The highest levels of thyroid contamination were recorded in the technician just prior to the end of the usage of liquid ^{131}I. The increase in values was mainly due to the need for frequent repairs and manipulation of the pipette within the laminar box before completely shutting down the device.

The most significant reduction of the thyroid exposure occurred in one of the radiological assistants (RA3). The mean value of measured activity decreased by 70 %. Higher values of activity measured in the thyroid compared to other radiological assistants in this case were due to less experienced manipulation of radiopharmaceuticals. The values of average thyroid radioiodine activities for individual workers are shown in Table 3.

TABLE 3. AVERAGE VALUES OF RADIOIODINE ACTIVITY IN THE THYROID GLAND FOR DNME WORKERS (RT – TECHNICIAN, RA1-4 - RADIOLOGICAL ASSISTANT 1-4) IN PERIOD 9/2012-08/2014.

Worker	Average activity in thyroid [Bq]		Decrease of measured activity (liquid to capsule) [%]
	Liquid	Capsule	
RT	46 (±43)	3 (±8)	93%
RA1	59 (±28)	31 (±19)	43%
RA2	92 (±54)	-	-
RA3	153 (±85)	46 (±38)	70%
RA4	-	4 (±2)	-

The results of personal dosimetry show that after administration of radioiodine in the form of capsules there was a significant decrease in the radiation exposure from the external irradiation of the workers concerned. The average monthly value of the personal dose equivalent $H_p(10)$ decreased on average by more than 70 % (from an average value of 0.36 ± 0.06 mSv to 0.10 ± 0.05 mSv) and a personal dose equivalent for the evaluation of doses on the skin and the limbs $H_p(0.07)$ decreased on average by 65 %. Data on radiation from finger-based TLD dosimeters revealed a reduction in the annual equivalent dose on skin $H_{(T)}$ by more than 75 %. To illustrate the decrease of the radiation exposure, the average results of whole-body dosimetry were plotted in Fig. 1 during the monitored periods. As with the accumulation of activity in the thyroid, until the end of the pipette operation, the dose in the technician did not increase. Since the introduction of capsules, it is seldom that the values on the personal dosimeters of the selected workers are measurable. In the case of $H_p(10)$, they do not exceed 0,05 mSv.

4. DISCUSSION

Measured levels of thyroid contamination are lower than those reported by other available articles [3–4]. Compared to the measurements performed by G. Krajewska and K.A. Pachockim; 2013, our measured thyroid contamination values are significantly lower both when administering ^{131}I liquid and when administering capsules. In his article, the thyroid contamination values of medical staff at the ^{131}I nuclear medicine department exceed 280 Bq. For the technician, the average value was 83 Bq. Monthly DNME's personal dose equivalents range from hundreds of limits for radiation workers. Since neither the limits nor the original optimization limits of the workplace were exceeded when using radioiodine in the form of a liquid, the combination of both dosage forms could be a suitable variant for the administration of ^{131}I. In the current knowledge of the authors, there is

no relevant study to address the radiological exposure of workers when passing radioiodine therapy from liquid to capsule.

From the point of view of the financial cost of radiopharmaceuticals in different dosage forms, it is financially advantageous to administer activity as a liquid in diagnostic applications. The cost of liquid radioiodine, converted to 1 MBq (for individual patient activities up to 200 MBq), is currently approximately 15 times lower than the same amount of capsule activity. With increasing activity, however, the cost of 1 MBq of capsules decreases, and for the commonly used therapeutic activities of the radioiodine, the price of solution and capsules is only slightly different.

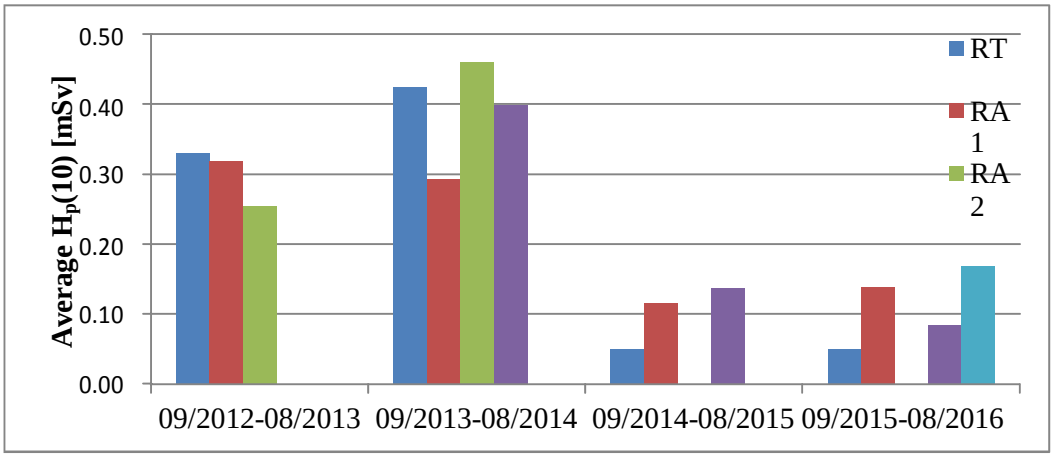

FIG. 1. Summary of average values of Hp (10) for DNME workers in the period from September 2012 to August 2016 (RT - technician, RA1-4 - radiological assistant 1-4).

5. CONCLUSION

The published work offers a unique comparison of thyroid contamination measurements and personal dosimetry results of the personnel involved in the preparation of the radioiodine administered to patients in diagnostic and therapeutic applications. The results suggest that the choice of radiopharmaceutical dosage form may affect the magnitude of the radiation exposure of the most exposed individuals, but the absolute value of the saved radiation exposure is not large. The decrease in the radiation exposure from both internal and external irradiation was tens of percent in the workers surveyed when switching from liquid to capsules. However, the collective exposure of workers is only 0,01 manSv/year. Thus, it is apparent that such procedures for working with the liquid form of radioiodine were well optimized. The decrease in $H_p(10)$ was significant in the transition to capsules but due to the optimization limits in legislation (decree on radiation protection no. 422/2016 Coll.), the positive effect is substantiallyreduced.

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IMPLEMENTATION OF THE DOSE CONSTRAINT CONCEPT IN OPTIMIZATION OF OCCUPATIONAL EXPOSURE IN NUCLEAR MEDICINE SECTOR IN POLAND

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Abstract

The implementation of the ICRP principle of the process of constrained optimisations even for occupational exposures still hit on methodology problems in practice. This principle implies requirement of new assessments to ensure that "the likelihood of incurring exposure, the num-ber of people exposed and the magnitude of their individual doses are kept as low as reasonably achievable.

A case study for dose constrain values for nuclear medicine stuff of endocrinology units has been performed. The external exposure has been evaluated by temporary measurements with passive TLD. The internal contamination was measured by portable NaI spectrometer. The envi-ronmental conditions, routine procedures and number of treated patients and administrated ac-tivity of I-131 were recorded. Base on the data MC generated PHP's and the 95% confidence tail of the doses for the most highly exposed individuals were calculated with ORACLE CRISTALL BALL softwere.

The study has revealed, that PHP's shapes had unregular shapes, depending on random oper-ations rather than routine procedures. Analysis of data statistics showed need of adequate nu-merous monitoring data of internal contamination. Due to accidental characteristics of monitor-ing data, the 95% confidence tail of the doses for the most highly exposed individuals in the en-docrinology units is near the limit of 20 mSv per year.

1. INTRODUCTION

In the last, revised ICRP Publication 103 [1], idea of dose constraint implementation in all exposure situations is conceptually addressed through the process of constrained optimisations. The revisions of the International Basic Safety Standards [2], and the Euratom Basic Safety Standards Directive [3] both aim to implement new ICRP recommendations and have requirements to use dose constraints, defined broadly along the lines provided by the ICRP. These will be obligatory adopted in the national regulations by regulatory authorities of EU countries. This principle implies requirement of new assessments to ensure that "the likelihood of incurring exposure, the number of people exposed, and the magnitude of their individual doses are kept as low as reasonably achievable". The role of dose constraints for occupational exposures, was previously elaborated in ICRP Publication 101b [4] in the statement "the dose constraint is a value of individual dose used to limit the range of options considered in the process of optimisation". The practical implementation of this ICRP principle in the process of constrained optimisations for occupational exposures in nuclear medicine units still hit on methodology problems in practice. Indeed, due to sparse data of individual dose records, particularly a lack of internal exposures assessments as well as random characteristics of monitoring data, the occupational dose constraints use to be arbitrary set near the limit of 20 mSv per year. This case study demonstrates that for the particular endocrinology procedures dealing with ^{131}I post-surgery therapy, the 90% confidence tail of the doses for the most highly exposed individuals exceeded the limit of 20 mSv per year. One might concluded that the ALARA principal target of occupational exposures management can be reasonably achieved when supported by numerous and high quality personal doses data.

2. MATERIALS AND METHODS

The aim of this case study of dose constrain values for nuclear medicine staff of endocrinology units was to set up the baseline of existing doses together with statistical distribution. The external exposure has been evaluated both by temporary measurements with passive TLD carried by stuff as well as calculated using

dosimetry model. The internal exposure due to inhalation of airborne ^{131}I (exhaled by patients) was evaluated based on experimental determined ratio AC-AD (average ^{131}I air concentration in patient's room to administered dose of ^{131}I). The environmental conditions, number of treated patients and details of routine procedures as average distance from patient, contact time with patient as well as administrated activity of ^{131}I have been considered. Base on the data Monte Carlo generated PHP's and the 95% confidence tail of the doses for the most highly exposed individuals were calculated with ORACLE CRISTALL BALL software.

2.1. Determination of ratio AC-AD (average ^{131}I air concentration in patient's room to administered dose of ^{131}I)

Average air activity of ^{131}I concentration in patients' room (volume of 25 m^3), was determined by series of hourly direct measurements with mobile sampling station (see Fig. 1a). The air flow rate was (40 m^3h^{-1}), filter consists of three layer of charcoal impregnated TEDA filter (1,5-2%) with KI (1,5-2%). The filtration efficiency was estimated as 99.9%. Average measured ^{131}I activity concentration in air was about 100 Bqm^{-3}. Conversion ratio AC-AD was evaluated by fitting predicted air concentrations to average measured air concentrations (see Fig. 1b). Value of AC-AD was equal to: $1.7\times10^{-8}[m^{-3}]$

2.2. Validation of external dosimetry model

To check the external exposure model the set of environmental TLD were exposed in selected places of two patient's rooms i.e.: about 3m from patient's bed, in aisle, patient's toilets and patients ante-chamber. The exposition time was 100 days, then the average dose rate was calculated. The error of calculation was estimated about 50%. The dose rate were predicted by external exposure model taking in to account administered activity to patient (1GBq), distance, scattered radiation and wall shielding. Comparison of dose rate predicted by dosimetry models and estimated by TLD passive dosimetry is presented on Fig. 2.

2.3. Monte Carlo simulation

Dosimetric model due to its simplification of complex phenomena enables only to calculate dose with significant range of uncertainty. MC simulation allows to reveal the range and probability of occurrence predicted doses values. For this purpose, the critical models parameters as exhalation ratio to air by radioiodine administrated patient, the contact time with patient by medical staff (doctor, nurse), the distance from patient to medical staff during investigation as well as number of performed particular treatments were set as min, max range with predefined shape of probability. The bound of the particular ranges was determine as a result of interviews with medical staff. The example of selected parameters of dosimetric model and resulting predicted doses are presented in Table 1 for the most hazardous procedures.

3. RESULTS AND CONCLUSIONS

The dosimetric model for purpose to evaluate occupational hazard in nuclear medicine endocrinology laboratories using ^{99m}Tc and ^{131}I for diagnostic and therapy of thyroid diseases has been evaluated an validated experimentally.

Ten nuclear medicine procedures (eight procedures using pharmaceutics with ^{131}I and two procedures with ^{99m}Tc), considered as sources of high occupational hazard were investigated.

Sensivity analysis of dosimetric model of occupationally exposed personnel revealed the most critical parameters that might yield to the highest doses and should be taken in to account in the process of constrained optimisation. These are: distance from patient and contact time with patient. The third parameter i.e. the number of investigations per year that might change of about 30% in consecutive years had less influence on personnel yearly doses.

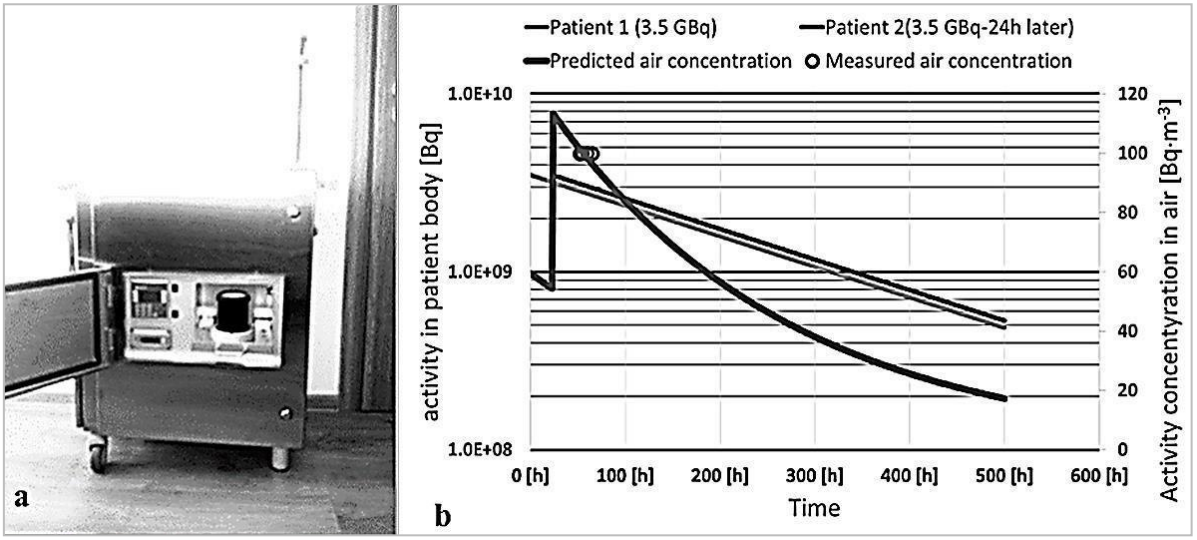
The exhalation rate of radioisotope of ^{131}I by patients and ventilation conditions in the patients' room has minor effect on total doses because contribution of internal exposure to the total dose constitutes about 1 percent.

The highest occupational hazard to involved personnel, especially to nurses might be related with one therapeutic procedure "*supplementary treatment of thyroid cancer with* ^{131}I" because of high activity administrated (1 GBq) and seven investigations periods during the treatment. Moreover two other diagnostic procedures i.e. "*post treatment of whole body scintigraphy*" and "*post treatment of whole body scintigraphy with MIBG*" due to high activity administrated (1.5 GBq, 1 GBq respectively) also yields to higher occupational risk. Special care and action should be taken to keep doses low by limiting distance and contact time with patient. Additionally, extra shielding should be considered.

TABLE 1. EXAMPLE OF OPTIMISATION USING MONTE CARLO ANALYSIS FOR HYPERTHYROID TREATMENT PROCEDURE

Procedure description:
administered dose: Complex:(sodium iodine NaI); activity of 131I (1 GBq)
contact with patient: 7 investigations during the treatment at: 2, 12, 24,36, 48, 54, 60 [h]
Number of treatments per year:
average : 100; range: 80÷120; PDF: triangular
Ratio AC-AD: central value: 1.7×10-8 , range: 5.7×10-9÷5.1×10-8; PDF: uniform

Staff	Contact time with patient1 [h]	Distance from patient [m]	Distribution of doses [mSv·year^{-1}]		
Doctor	average: 0.1	average: 0.75	Internal exp.:	mean: 1.1×10^{-1}	(median: 1.0×10^{-1})
			min: 1.5×10^{-2}	max: 2.7×10^{-1}	90% tail < 1.8×10^{-1}
	range: 0.05 ÷0.15	range :0.25 ÷1.0	External exp.:	mean: 18.4	(median: 17.6)
			min: 6.6	max: 55.4	90% tail < 25.3
	PDF: triangular	PDF: triangular	Total exp.	mean: 18.5	(median: 17.5)
			min: 6.6	max: 55.5	90% tail < 25.5
Nurse	average: 0.1	average: 0.5	Internal exp.:	mean: 5.2×10-1	(median: 5.1×10-1)
			min: 6.1×10-2	max: 1.5	90% tail < 8.7×10-1
	range: 0.05 ÷0.15	range:0.20 ÷1.0	External exp.:	mean: 27.4	(median: 26.1)
			min: 8.2	max: 81.4	90% tail < 38.6
	PDF: triangular	PDF: triangular	Total:	mean: 27.9	(median: 26.6)
			min: 8.9	max: 81.9	90% tail < 39.1



[1] The same for 7 periods of treatment

FIG. 1. Mobile sampling station for measurement of airborne ^{131}I (a). Determination of conversion ratio AC-AD - average 131I air concentration in patient's room to administered dose of ^{131}I (b)

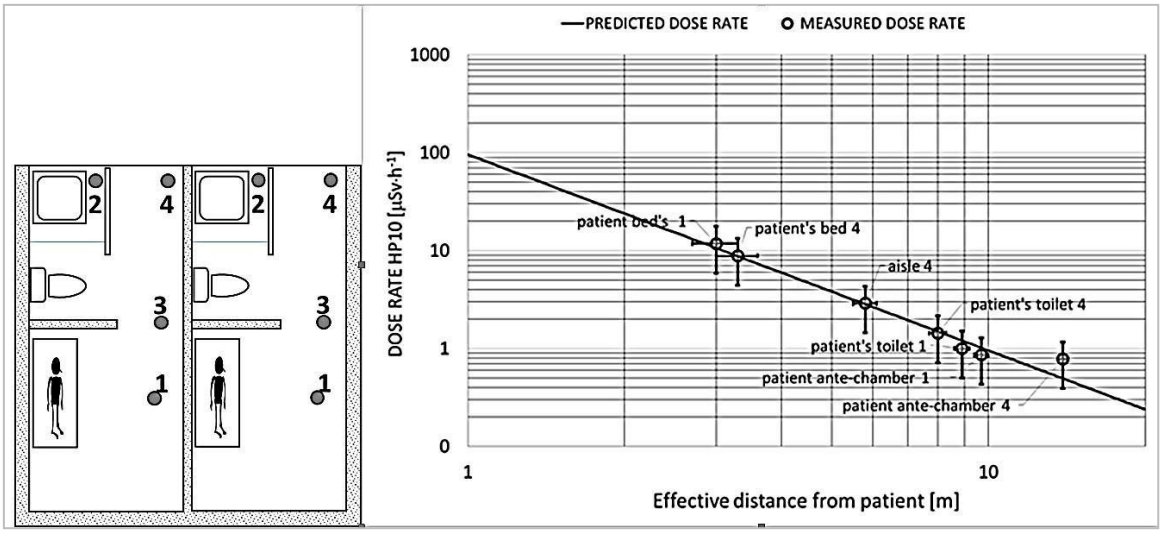


FIG. 2. Placement of environmental TLD: 1: bed, 2:toilet, 3: aisle, 4: ante-chamber. Comparison of dose rate predicted by dosimetry models and estimated by TLD passive dosimetry.

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ASSESSMENT OF RADIATION DOSE NEARBY PATIENT ADMINISTRATED WITH ^{123}I FOR THYROID SCINTIGRAPHY

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Abstract

Whole body diagnostic scanning (WBS) with specific dose of radioiodine ^{123}I is performed routinely as a post-surgical imaging to evaluate the extent of thyroid cancer. To assess the nearby patient radiation exposure, 112 WBS patients were monitored and their external dose rate at the thyroid level at distances of 5 and 50 cm were measured after10 minutes from dose administration. The patients were classified into two groups according to patient sex. The external dose rate for female patient group was in the range of 18.2-52.4 and 4.45-20.0 μSv.h^{-1} respectively, while it was ranged in 15-50 and 2.90-13.20 μSv.h^{-1} for male patient group. Two realistic scenarios of medical staff exposure were proposed: 5 minutes for close contact at a distance of 5 cm from patient, and 20 minutes at a distance of 50 cm. The maximum calculated dose received by a medical staff for the both scenarios was 4.37 and 6.67 μSv correspondingly. In order to maintain conformity with the radiation protection concepts, and to optimize the occupational exposure, medical staff are advised to optimize the contact with WBS patient in time and distance.

1. INTRODUCTION

As a thyroid scanning agent, radioiodine ^{123}I represents the most suitable isotope of iodine for the diagnostic study of thyroid diseases [1]. It has a suitable half-life of 13.3 h for the 24-h iodine uptake test and a pure gamma emission of 159 keV; ideal for NaI crystal detector of current gamma cameras. As compared with radioiodine ^{131}I, it also has much greater photon flux, which gives approximately 20 times counting rate for the same administered dose [2]. Whole body diagnostic scanning (WBS) with radioiodine ^{123}I is performed routinely as a post-surgical imaging to assess the extent of thyroid cancer, especially distant metastasis prior to the therapeutic dose administering of ^{131}I especially in patients with Differentiated Thyroid Carcinoma (DTC) [3]. A specific dose of ^{123}I between 111 and 185 MBq (3 to 5 mCi) is administrated to patient under withdrawal conditions in order to obtain a whole body scan images primarily at 6 and 24 hours post dose administration [4-6] and the patient is released from the hospital without any isolation period.

From the radiation protection point of view, in spite of the low administrated radioiodine dose, patients undergoing whole body scan may subsequently come into close contact with members of public and hospital staff. In fact, the WBS procedure requests a delayed imaging after 6 and 24 hours post dose administration hence the patient is released from hospital without any virtual isolation period after the dose administration. In the scientific literatures, several research papers and guidelines reports were focused on the external dose rate from patient treated by radioiodine 131 in discussing the release criteria from hospital [7,8] or assessing the related public exposure [9,10]. Other research papers were discussed the radiation dose to the surrounding from patients who are undergoing nuclear medicine examination using radiopharmaceuticals such as ^{111}In and ^{99m}Tc [11] and ^{201}Tl and ^{99m}Tc [12].

The goal of the present work is to assess the radiation dose in positions nearby a patient administrated by radioiodine ^{123}I in order to optimise the radiation protection of occupational exposure.

2. MATERIAL AND METHOD

In this study, a total of 112 patients administrated with ^{123}I for whole body scan at AL-BAYROUNI University Hospital (Damascus City) were monitored from August to December 2016. The patients were selected randomly. The administered activities ranged from 111 to 185 MBq (3–5 mCi). The WBS is effectuated 6 and 24 hours post ^{123}I administration. The patients stayed in a special room in the hospital before the first whole body scanning, then they released from hospital for the next day for the WBS 24 hours post dose administration.

The universal survey meter type RADOS-200 has been used to measure the external radiation dose rate from patients administrated ^{123}I. The measurement range of this system is from 0.01 $\mu Sv.h^{-1}$ to 10 $Sv.h^{-1}$, which is suitable to measure dose rate in ^{123}I patient. This survey meter is also calibrated annually in the National Radiation Metrology Laboratory (NRML) in Syria. The uncertainty associated with survey measurements was provided by the calibration procedure and was better than 5 %. Measurement of the patient external dose rate was performed at two distances of 10 and 50 cm from the effective point of measurement and standing point of the patient for the thyroid level after 10 minutes of dose administration. The patients sample was divided into two groups according to patient sex. For all sample patient, the age, height, and weight were recorded. Table 1 represents the anthropometric measurements of patients groups. The illustrative statistical methods, average, standard deviation, were used during the analysis of study data.

TABLE 1. Anthropometric measurements of patient groups (Mean±SD)

	Female	Male
No. of patients	93	19
Age (years)	44.40±12.07	45.16±2.62
BMI (kg/m^2)	30.54±6.00	29.85±1.25
Administrated dose (MBq)	166.30±18.25	167.47±3.57

3. RESULTS

The mean values of the measured external radiation dose rates and the normalised dose rate in terms of administered activity for female and male patient groups at distances of 5 and 50 cm are presented in Table 2. After 10 minutes from patient dose administration, mean of dose rate in female patient group at distance of 5 and 50 cm was within the range of 18.2-52.4 and 4.45-20.0 $\mu Sv.h^{-1}$ respectively, while the mean of dose rate in male patient group was in the range of 15-50 and 2.90-13.20 $\mu Sv.h^{-1}$ correspondingly. Furthermore, regardless the distance from the measurement point, the difference in the average of external dose rate between the female and male patients' groups was insignificance.

TABLE 2. External dose rate ($\mu Sv.h^{-1}$) and normalised dose rate in terms of administered activity ($\mu Sv.h^{-1}.MBq^{-1}$) at distances of 5 and 50 cm from patient (Mean±SD)

Patient Group	At 5 cm		At 50 cm	
	External dose rate	Normalized dose rate	External dose rate	Normalized dose rate
Female	32.131±6.786	0.195±0.043	8.667±2.492	0.052±0.015
Male	33.384±2.330	0.201±0.015	8.438±0.619	0.050±0.004

Box plots of external dose rate from female and male patients' groups after 10 minutes at 5 and 50 cm are shown in Fig 1 and Fig 2 respectively. In this study, P value of less than 0.0001 was considered significant for the measured external dose rates of both patient groups at the confidence level of 95%.

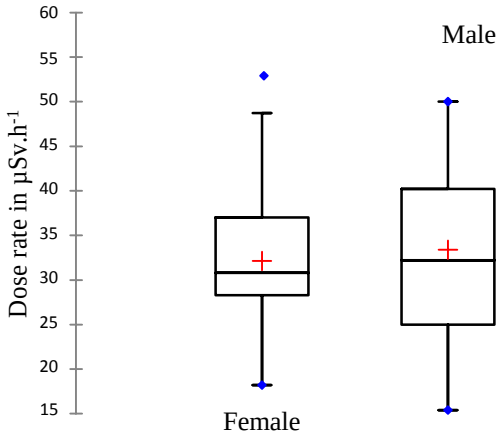

FIG. 1. Dose rate from patient at 5cm distance after 10 minutes of ^{123}I dose administration

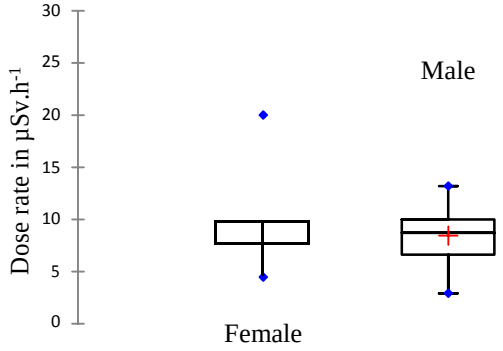

FIG. 2. Dose rate from patient at 50 cm distance after 10 minutes of ^{123}I dose administration

4. DISCUSSIONS

After dose administration, patients become a source of exposure to their relatives, medical staff and other people who are in close contact with them. This exposure is related primarily to the distance from patient and exposure duration and also to the administered dose. The maximum external dose rate from patient measured at 5 and 50 cm was 52.2 and 20 $\mu Sv.h^{-1}$ respectively.

In order to estimate the radiation dose received by a medical staff nearby a WBS ^{123}I administrated patient, two realistic scenarios simulating the feasible exposure duration were proposed as 5 minutes for close contact at a distance of 5 cm from patient and 20 minutes for contact at 50 cm. The effective dose at a specific distance after the desired duration was calculated from the related dose rate measurements. For the first scenario, the maximum radiation dose received by an accompanying medical staff was 4.37 µSv (the mean value is 2.71±0.64) while for the second scenario, this value was 6.67 µSv (the mean value is 2.88±0.84) at 50 cm from patients.

According to International Commission on Radiological Protection (ICRP), the occupational exposure dose is limited to 20 mSv annually [13], consequently, the derived medical staff occupational dose is limited to 10 µSv per hour which includes the received dose from administrated patients and other radiation sources in nuclear medicine environments such as syringe and vials. Moreover, the International Agency of Atomic Energy estimates that the average annual monitored effective dose for exposed workers in a nuclear medicine facility is ranged from 3 to 5 mSv [14]. It is obvious that the hourly dose limit of a medical staff can be achieved by simply exposing to two WBS ^{123}I administrated patients for 5 minutes from a distance of 5 cm or for 20 minutes at a distance of 50 cm. The estimated annual effective dose of a medical staff member could be excessed the occupational dose limit for an average workload consists of 5 WBS ^{123}I patients daily. Therefore, the contact with WBS ^{123}I administrated patients in the waiting room and through the patient preparation to the whole body scan should be optimized in distance and duration in order to maintain the conformity with radiation protection ALARA principles and minimize the resulted occupational exposure.

5. CONCLUSION

The radiation dose nearby a patient administrated with ^{123}I for whole body scanning has been assessed. The external radiation dose rate from 112 patients at 5 and 50 cm was measured. Two realistic scenarios consisting a close contact distance from patient at 5 cm for 5 minutes and a contact at 50 cm for 20 minutes were proposed to determine the maximum received medical staff effective dose. According to our findings, the hourly dose limit of occupational exposure can be achieved by simply exposing to two ^{123}I administrated patients for 5 minutes from a distance of 5 cm or for 20 minutes at a distance of 50 cm. So the medical staff should optimize the close contact with patient in time and distance in order to minimize the related occupational exposure.

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IODINE 131 STAFF CONTAMINATION IN NUCLEAR MEDICINE DEPARTMENT

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Abstract

To prove internal Iodine 131 (^{131}I) contamination of nuclear medicine (NM) staff by external thyroid activity counting and ^{131}I urinary excretion measures.
Thyroid activity (internal contamination) was measured for the staff of NM department of the military hospital of Tunis, which includes 12 workers.
A 2x2 inch solid NaCl crystal detector without collimator (Camberra SG-2R probe) combined to colibri Camberra TTC monitoring system was used. Urinary activity was measured, using a CAPRAC-t® counter, by collecting fresh urine for nine workers and 24h collected urine for the 3 others.
Ten GBq ^{131}I is usually used (1,85 GBq in solution form). The mean background activity was around 85 cps. The thyroid activity was ranging from 95 to 120 cps in eleven workers. One technician had high thyroid counting reaching 174 cps. All urine samples were equal to background activity except for the same one technician who had a 364Kev peak over the background spectrum. These findings show evidence of internal contamination of one technician by over exposure to ^{131}I in solution form.
Iodine 131 contamination is a hazard that should be taken in account while manipulating solution forms and should be monitored by thyroid and urine activity measurements regularly.

1. INTRODUCTION

Nuclear medicine is a well-developed field for imaging and treating patients. Iodine-131 is considered one of the most frequently used radionuclides for diagnosis and radiotherapy of thyroid diseases [1]. The treatment consists mostly on radioiodine 131 therapy for either differentiated thyroid cancer (DTC) or benign hyperthyroidism. It had been performed on patients and has contributed to normalization of life expectancy in the cases of DTC and treating and improving the life quality for patients with hyperthyroidism [2]. Nevertheless, manipulating such radioactive product require safety rules because of the risks of workers contamination. Internal intakes can be determined by direct and indirect measurement methods.

The aim of our study is to prove internal Iodine 131 contamination of nuclear medicine staff by external thyroid activity counting and 131 Iodine urinary excretion measures.

2. METHODS

Thyroid activity, resulting from internal contamination, was measured for the whole staff of nuclear medicine department of the military hospital of Tunis. The stuff included 12 workers. It was composed of five doctors, four technicians and three administrative agents. For that purpose, we proceeded to a direct measurement by detecting emitted photons from internally deposited radionuclides in the thyroid gland. Biological internal contamination was measured in urines samples and was considered as an indirect method.

The workers did not have the same exposure. Technicians were directly manipulating the radioiodine131. Doctors had less contact with this product. Administrative agents did not have any contact with the radioiodine 131.

The measurements were done twice for each worker, in two consecutive usual working days. All the workers were present in the nuclear medicine department during the six days before the measurements and were performing their usual daily activities. No one had any rest day during that period. The worker who benefited from a rest day was excluded from the study in order to keep similar conditions for all.

For that purpose, a 2x2 inch solid NaCl crystal detector without collimator (Camberra SG-2R probe) combined to colibri Camberra TTC monitoring system was used.

First, background activity was measured during ten minutes. Then we proceeded to the external thyroid activity counting during ten minutes. The result of the thyroid measures, from which we removed the background activity, was the effective external contamination considered.

Urinary activity was measured by collecting fresh urine samples for nine workers and 24h collected urine for the 3 others. A CAPRAC-t® counter was used for that purpose, during ten minutes per measure.

We considered the activity and the spectrum given by the CAPRAC-t® counter.

3. RESULTS

In our department, 10 GBq Iodine 131 is usually used (1,85 GBq in solution form).
The mean background activity was around 85 cps, in the several area of the department, excluding the laboratory which background activity's was higher. The background was about 0.7 µSv in this area and about 1.22 µSv behind the leaded panel in the acquisition room.
The thyroid activity was ranging from 95 to 120 cps in eleven workers, despite of their various tasks either they manipulate the radioiodine 131 or not. Only one technician had high thyroid counting reaching 174 cps in both measures. The measures are presented in the table below (Table1).

Workers	Thyroid Activity Measured (cpm)
Doctor 1	118
Doctor 2	110
Doctor 3	99
Doctor 4	98
Doctor 5	101
Technician 1	116
Technician 2	120
Technician 3	113
Technician 4	174
Administrative agent 1	95
Administrative agent 2	97
Administrative agent 3	108

Table 1 : Thyroid Activity measurments

When the urine measurements,the mean background activity was about 541cpm (Figure 1). All urine samples were equal to background activity except for the same one technician for whom it was about 727cpm. He had two additional peaks over the background spectrum: a 364Kev (radioiodine) and a 140 Kev (Technetium 99m: ^{99m}Tc) peaks (Figures 2,3,4).

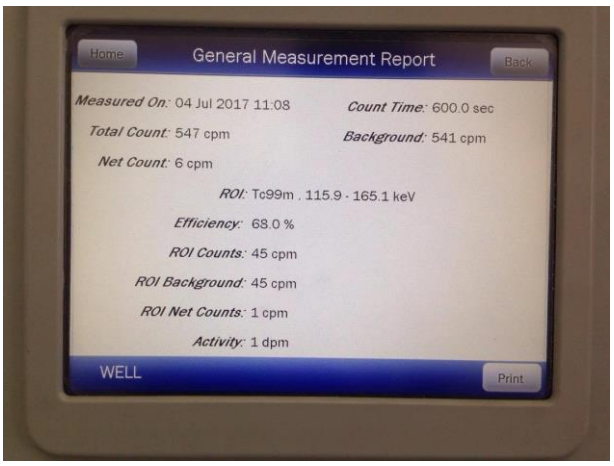

FIG1 : Mean background activity

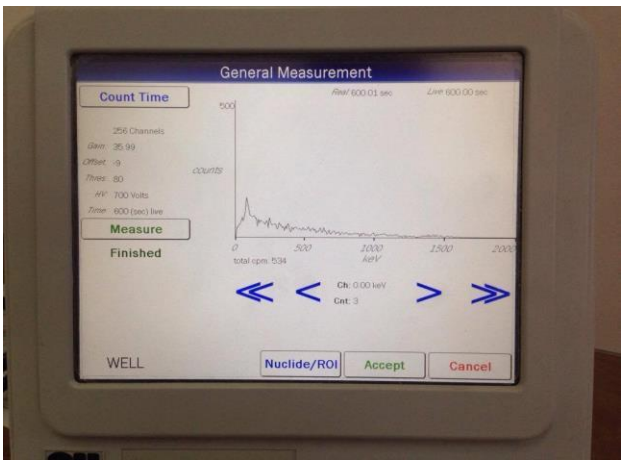

FIG2 : Average spectrum for the eleven workers

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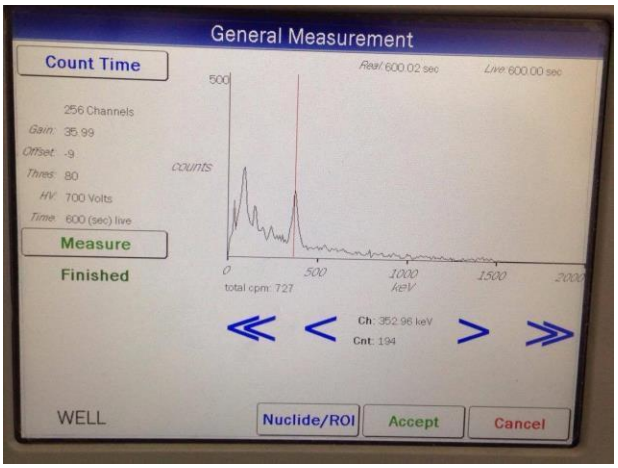

FIG3 : Radioiodine^{131} peak

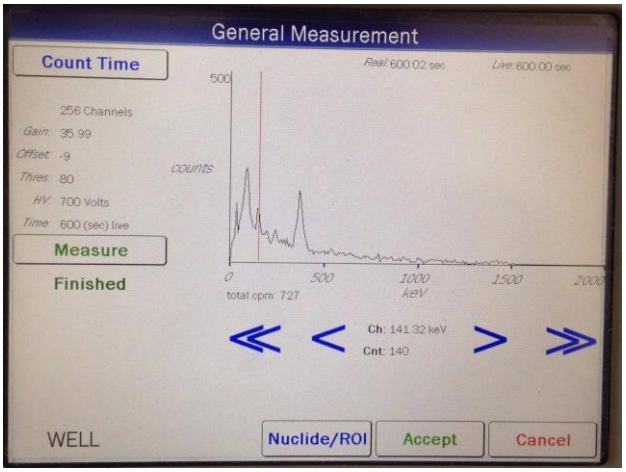

FIG4 : ^{99m}Tc peak

These findings show evidence of internal contamination of one technician by over exposure to Iodine 131 in solution form.

4. DISCUSSIONS

The background measurement result was similar to the study of Ghezini [3] which was equal to 0,6 µSv.
Eleven workers had nearly equal thyroid activity measurements and urine activity despite their different tasks and roles in the department: the technician, who manipulates most the radioiodine 131, has similar measurements as the administrative agent. That indicates that radioprotection rules are respected and applied.
For one worker, the thyroid activity measures were higher than the standards and comparing to the rest of the team. It was confirmed by two measures. The analysis of the urine samples activity spectrum showed two additional peaks comparing to the rest of the nuclear medicine department stuff: a 364 Kev peak and a 140 Kev peak.

These respectively corresponds to a radioiodine 131 peak and to ^{99m}Tc, which are the most used radionuclides in our department. This suggests a disrespect of radioprotection protocols. Moreover, manipulating radioiodine in solution form, which is volatile, exposes to internal contamination by inhalation more than any other form. The inhalation of radioactive airborne particles is one of the most important routes of entry of radionuclides into the human body [4]. That is a reason for strict radiation protection in nuclear medicine departments. So, liquid form of radioiodine should be avoided as much as possible to protect workers from the risk of inhalation [4].
For the benefit and protection of the stuff of nuclear medicine departments, standards of radiation protection should be respected at every step. Workers and locals should be monitored, and quality assurance programme are to be adopted [5].

1. CONCLUSIONS

Iodine 131 contamination is a hazard that should be taken in account while manipulating solution forms and should be monitored by thyroid and urine activity measurements regularly. They are reliable means of assessment. In our case, such assessments are to be done frequently. That one worker with higher internal contamination should be monitored individually and should be observed in order to detect and correct any radiation protection disrespect.

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EYE DOSE ASSESSMENT OF STAFF IN NUCLEAR MEDICINE AND RADIOPHARMACY DEPARTMENTS

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Abstract

The Basic Safety Standards Directive issued in December 2013 has reduced the eye lens dose limit to 20 mSv annually, from 150 mSv that it is today. It is still unknown what impact this change will have in hospitals and whether some staff will require additional monitoring of their eye doses or whether they will need to become classified workers. This study aimed to assess the impact of the new eye lens dose limit on our Nuclear Medicine and Radiopharmacy staff. In particular, it looked to assess the need for regular monitoring, additional shielding requirements and also to identify staff likely to become classified workers. Various members of staff in the two departments were issued with the EYE-D holder containing Thermoluminesence Dosimeters in order to measure the eye dose they receive. The results showed that members of staff in the two departments are likely to receive around 3 mSv of eye dose annually, based on their current workload. This is well below the new eye dose limit and the classification level and indicates that the current shielding measures in place are sufficient. Periodic eye dose monitoring may be necessary though to ensure the doses are kept at low levels.

1. INTRODUCTION

Recent epidemiological studies performed with improved dosimetric techniques showed an increased number of radiation induced eye cataracts at low doses [1]. Based on these studies, the International Commission on Radiological Protection (ICRP) concluded that the threshold dose for eye cataract is approximately 0.5 Gy for acute exposures, compared to the 5 Gy that it was assumed to be until recently (ICRP Publication 118) [2].

Following these results, ICRP recommended in 2011 that the eye lens dose limit should be reduced to 20 mSv a year from 150 mSv that it is today. The European Union (EU) implemented this recommendation in 2013 and published a new Basic Safety Standards Directive (2013/59/Euratom) [3]. Article 9 (Dose limits for occupational exposure), paragraph 3 of this Directive says:

"In addition to the limits on effective dose laid down in paragraph 2, the following limits on equivalent dose shall apply:

(a) the limit on the equivalent dose for the lens of the eye shall be 20 mSv in a single year or 100 mSv in any five consecutive years subject to a maximum dose of 50 mSv in a single year, as specified in national legislation"

Member states of the European Union (EU) have a deadline until February 2018 in order to implement this Directive and issue national legislation.

The reduction of the eye lens dose limit is significant and is expected to affect many sectors where ionising radiation is used, including the medical one [4]. Risk assessments need to be performed in various areas of work with ionising radiation in order to assess the implications and identify those members of staff who are likely to exceed the new eye dose limit or the proposed classification level of 15 mSv.

To this end, a study was performed aiming to measure eye doses for staff working with ionising radiation at Colchester Hospital University NHS Foundation Trust in the UK. This report presents the results of the eye dose measurements in the Nuclear Medicine and Radiopharmacy departments of our hospital. The aim of this study is:

— To measure eye doses to staff working with ionising radiation in Nuclear Medicine and Radiopharmacy;
— To assess the need for regular monitoring;
— To assess the need for additional shielding (protective equipment);
— To identify staff likely to exceed the proposed classification level of 15 mSv.

2. METHODS

2.1. Eye lens dose assessment method

Eye lens dose is not directly measurable. Hp(3) is a quantity used for measuring the eye dose and is defined as the equivalent dose at 3 mm depth where the sensitive tissue of the eye is located. However, Hp(3) is hardly used at present as there are very few dosimeters designed to measure Hp(3). Several methods have been proposed in the past for the estimation of the eye lens dose. Some of them use the dose measured by a dosimeter worn at the collar or waist level and correlate this dose to the dose on the eye lens. However, these methods involve high uncertainty based on the location where the dosimeter is worn, the staff's practice etc. Other methods suggest the use of published data, such as eye dose per procedure or eye dose per unit of activity (or per DAP in radiology) [5].

The use of a dosimeter worn as close as possible to the lens of the eye is considered to be the most accurate method for eye dosimetry. A dedicated eye lens dosimeter holder (EYE-D™, RADCARD) was developed as part of the EU "Optimization of radiation protection for medical staff" project (ORAMED) specifically to measure Hp(3) [6]. The EYE-D holder was used for this project. As shown in Fig. 1 below, EYE-D consists of a TLD pellet and a plastic (polymide) capsule where Thermoluminesence dosimeters (TLD) can be placed.

FIG. 1. The EYE-D holder with a TLD inside the capsule

Fig. 2 below demonstrates how the EYE-D dosimeter is used as close to the eye as possible, attached to a headband. Each member of staff can have two EYE-D dosimeter holders on their headband, measuring in this way the dose to the left and right eye simultaneously.

FIG. 2. The EYE-D dosimeter holder as close to the eye as possible, attached to a headband

2.2. Calibration of TLDs

The EYE-D holder was used with locally calibrated and read TLDs. The TLDs were calibrated by being irradiated on top of a 10 cm thick rectangular PMMA phantom. The calibration was performed at 195 kV with 1 mm of Cu in the beam using a Gulmay Orthovoltage unit. This gives an average energy of 140 kV. The TLDs were calibrated three times before use in order to ensure consistent and accurate calibration. They were read using a Harshaw 5500 TLD reader.

3. RESULTS AND DISCUSSION

3.1. Nuclear Medicine department

Four members of staff in the Nuclear Medicine department were assigned a headband with two EYE-D dosimeters attached to it, one for the left eye and one for the right eye. One member of staff was assigned a headband for four weeks. Three additional members of staff were assigned headbands for two weeks each. Annual dose was then estimated by extrapolating the measured doses. The four members of staff share the majority of the workload which includes injecting and scanning of patients. Staff 1 performs 25% of the workload, Staff 2 performs 25%, Staff 3 performs 25% and Staff 4 performs 20%. The workload of the department was estimated by analysing data from the last 3 years. An additional 10% workload was added to this to account for future changes. No I-131 therapies are performed currently in our hospital. Table 1 that follows shows the results from these measurements. All staff were wearing their headbands while injecting radiopharmaceuticals to the patients and during the patient scanning period as well.

TABLE 1. STAFF EYE DOSES IN THE NUCLEAR MEDICINE DEPARTMENT

Member of staff	Monitoring period	Eye	Measured dose (mSv)	Working weeks/year	Estimated annual dose (mSv)
1	4 weeks	L	0.249	50	3.11
		R	0.193		2.41
2	2 weeks	L	0.083	50	2.08
		R	0.045		1.12
3	2 weeks	L	0.058	50	1.45
		R	0.040		0.99
4	2 weeks	L	0.108	50	2.70
		R	0.068		1.70

It was noticed that during the monitoring period, Staff 1 performed more than the average number of exams. Also, the data for Staff 4 were excluded from the study as the headband was likely to have been unnecessary exposed (stored in the scanning room for a few days).

Looking at the doses in Table 1, it can be seen that all estimated annual eye doses are up to approximately 3 mSv. This is well below the new eye dose limit and these members of staff are not likely to exceed the proposed classification level under the current circumstances. Staff 1 that received the highest dose (3.11 mSv (L) and 2.41 mSv (R) annual doses) exceeded what is considered average workload during the monitoring period.

Variation in the dose delivered to the L and R eye was noticed for all staff. This is attributed mainly to the layout of the Nuclear Medicine rooms at our hospital and the way each member of staff stand with regard to the patient during the injection and scanning processes.

3.2. Radiopharmacy department

Four members of staff from the Radiopharmacy department participated in the study. They were given one headband with two EYE-D holders each for either one or two weeks. In order to avoid accumulation of high

doses, members of staff in Radiopharmacy follow a pattern of two weeks of work inside the unit and six weeks of work outside of the unit. Eye doses were monitored during the period inside the unit only. Based on our records of staff dosimetry, it can be assumed that the amount of dose staff receive while working outside of the unit is very low. Table 2 that follows shows the staff eye doses in Radiopharmacy.

TABLE 2. STAFF EYE DOSES IN THE RADIOPHARMACY DEPARTMENT

Member of staff	Monitoring period	Eye	Measured dose (mSv)	Working weeks/year	Estimated annual dose (mSv)
1	9 days	L	0.387	12	3.61
		R	0.240		2.24
2	1 week	L	0.082	12	0.98
		R	0.083		1.00
3	1 week	L	0.096	12	1.15
		R	0.082		0.98
4	1 week	L	0.061	12	0.73
		R	0.036		0.43

It can be seen that doses to staff in Radiopharmacy are low. Staff 1 had a spillage during the monitoring period and even if it assumed that there will be several spillages like that during the year, the eye doses do not exceed 3.6 mSv. Under normal working conditions including no spillages, the eye doses are around 1 mSv per year. These are well below the new dose limit and members of staff are not likely to reach the classification level of 15 mSv under the current circumstances. Also, small variations can be noticed in the dose received by the L and R eye, mainly attributed to working practices.

4. CONCLUSIONS

The results of this study have showed that the eye doses in our staff working in the Nuclear Medicine and Radiopharmacy departments are low and can be up to approximately 3 mSv per year, assuming a worst case scenario. These results are in agreement with other published studies in hospitals of similar size to ours [7]. No members of staff are expected to become classified workers in these two departments. Our members of staff are currently issued with ring TLDs (monitoring of equivalent dose to their fingers) and a body dosimeter (monitoring of whole body effective dose). It may be appropriate to issue them with a collar or headband dosimeter in the future. A periodic assessment of the eye doses at certain intervals or in cases of changes of equipment/workload is likely to be sufficient.

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Radioactive iodine -131 therapy for hyperthyroidism

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Abstract
Introduction: Hyperthyroidism is one of common pathological problem associated with thyroid gland. Radioactive iodine 131 therapy is uses for destroying cellular gland due to the high energy of beta emission.
Methods: Patients diagnosed with hyperthyroidism treated by Radioiodine 131during August 2016 and June 2017 at Radiation and isotopes center- Khartoum (RICK).
Results: Sixty patients 54 (90%) were female and 6 (10%) male. The mean and standard deviation of I 131 doses were 20.5± 8.6 with minimum dose is 10 mCi and the maximum dose is 30 mCi. Toxic goiter recorded high percentage among the types of thyroid disorders 29 (48.3%), and the second multi nodular goiter 14 (23.3%) followed by graves' disease 7 (11.7%), hyperthyroidism 6 (10%) and thyrotoxicosis 4 (6.7%).
Conclusion: Radioactive iodine 131 widely uses for hyperthyroidism treatment. The majority of the patients used radioiodine therapy in this study was female. Thus, protection is more important among the cases of pregnancy, breast-feeding and for children as well as public.

1. Introduction

Hyperthyroidism is one of common pathological problem associated with thyroid gland, in which thyroid hormone levels were increase [1]. Hyperthyroidism caused by graves' disease or by nodules within the gland that are locally overactive in producing too much thyroid hormone such as toxic multinodular goiter, and toxic adenoma [2].

1.1. Radioiodine (I131) therapy

Radioiodine therapy is a nuclear medicine treatment for an overactive thyroid; it is give orally on one time. I131 is use as the definitive treatment of choice in most patients with hyperthyroidism [3]; it is a choice for hyperthyroidism in any age [4]. The effectiveness of radioiodine treatment for hyperthyroidism is due to radiation-induced thyroid cellular damage resulting from high-energy beta emission. I-131 is absorbed into the bloodstream and concentrated by the thyroid gland, where it begins destroying the gland's cells [5]. In Sudan, found four centers are using radioiodine I131 therapy for hyperthyroidism as fellow: Radiation and isotopes center- Khartoum (RICK), Royal care hospital – Khartoum and Elneelain specialize center – Khartoum as well as one center in Shendi at north of Sudan [6]. In addition, precautions for patients who received radioiodine therapy proceed by the medical physicist in nuclear medicine department due to the legal guardian.

1.2. Radiation protection information for Patients after I131therapy

The instructions of Nuclear Regulatory Commission regulates some cautions necessary for protection after I131 therapy include , distance and time, it should be keep at a distance of about 2 meters and preferably avoid close contact to another people and keeping the time you are close to others to a minimum. Avoid close contact with pregnant woman and children; in addition, breast-feeding should be stopped. During restriction period, preferably sleep alone, flush the toilet, transportation are not long than 1 hour and avoid kissing because I131 release in Saliva [7].

The aim of this study to a chive the patients having increasing thyroid hormone levels were treated by radioiodine 131 in nuclear medicines, therefore, radiation safety that help obtain satisfactory clinical outcomes and optimize exposure to the patient and public.

2. Methods

Patients diagnosed with hyperthyroidism treated by a single dose of Radioiodine 131during August 2016 and June 2017 at Radiation and isotopes center- Khartoum (RICK). RICK is the first center using I131 in Sudan, and then followed by three centers.

2.1. Statistical analysis

SPSS Version 22 was used for descriptive the data of study as well as T. Test used for the mean and standard deviation of I131 doses.

3. Results

Sixty patients 54 (90%) were female and 6 (10%) male as in figure 1.

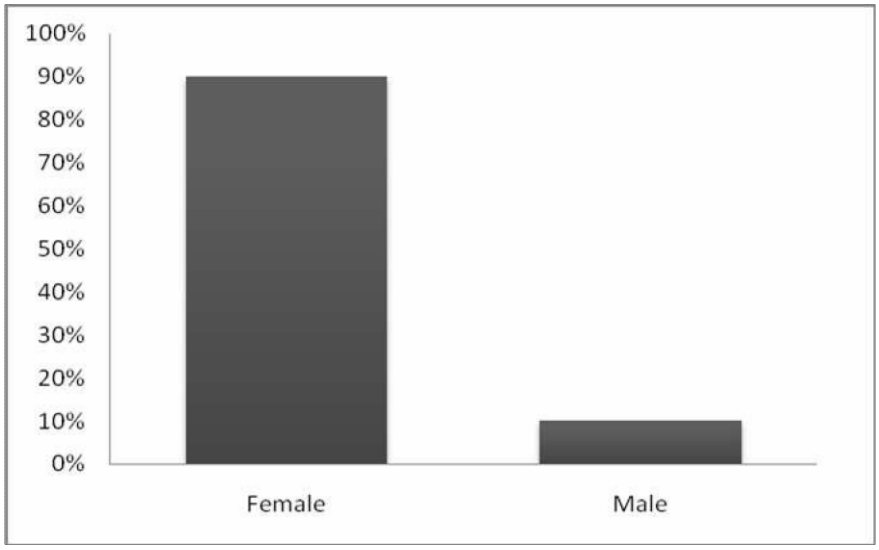

Fig 1: Distribution of patients according to sex

Toxic goiter recorded high percentage 29 (48.3%) among the diagnosis of overactive thyroid as explain in figure 2.

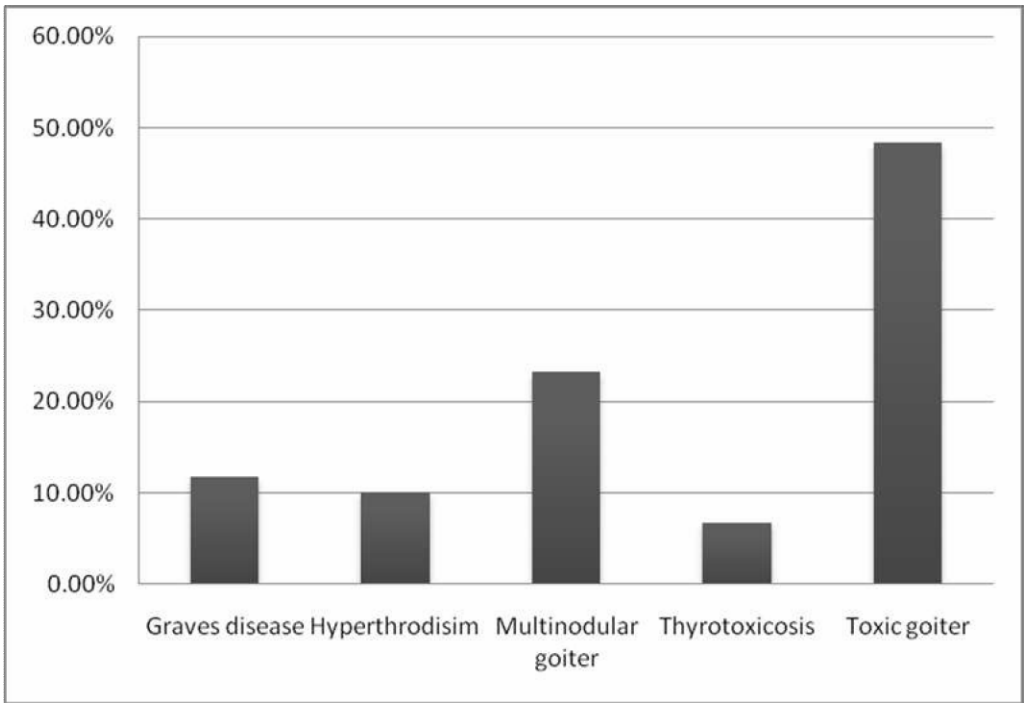

Fig 2: Diagnosis of thyroid disease for the patients treated by radioiodine 131

The mean and standard deviation of I 131 doses were 20.5 ± 8.6 with minimum dose is 10 mCi and the maximum dose is 30 mCi. Fixed radioiodine doses for the patients in different diagnosis received to 30 mCi except graves' disease it were from 10 to 18 mCi. Distribution of I131 doses by the patients according to the diagnosis of hyperthyroidism explained in table1.

Table 1: Shows I 131 doses among hyperthyroidism diagnosis for the patients in study

Diagnosis	I^{131} Doses / mCi									Total of diagnosed patients
	10	12	13	15	16	18	20	25	30	
Multinudular goiter	0	0	0	0	0	1	0	3	10	14
Toxic goiter	10	1	1	2	2	0	1	2	10	29
Hyperthyroidism	2	1	0	0	0	0	2	0	1	6
Graves disease	4	1	0	0	1	1	0	0	0	7
Thyrotoxicosis	0	0	0	0	1	0	1	0	2	4
Total of patients given I^{131}doses	16	3	1	2	4	2	4	5	23	60

4. Discussion:

From this study, it is clear that radioiodine 131 widely uses for hyperthyroidism treatment, because this data for 14 months from one centre in Khartoum, which confirms the increasingly used of I 131 as effective treatment of choice in most patients with hyperthyroidism [8]. Therefore, radiation protection satisfies the treatment outcomes, which confirmed by the study reported that after radioiodine (131I) therapy safely guides recommended for patients, their families and the public [9]. In present study, women used radioiodine 131 larger than men, which confirmed by the study showed that hyperthyroidism more common in women than men [10]. In this finding toxic goiter is the common types of hyperthyroidism, while, another study reported that, the most common pattern of thyroid disorders in Sudanese patients using nuclear medicine facilities in period between 2001- 2003 is simple multinodulargoiter [11]. Multinodular goiter is less common than graves' disease, but its prevalence increases with age and in the presence of iodine deficiency [12], while in this study, multinudular goiter is prelevace than graves' disease that may be due to the iodine deficiency as common problem in Sudan [13]. Also, our findings confirmed by the study showed that the most common causes of hyperthyroidism include toxic adenoma, toxic multinodular goiter and graves' disease [14]. I 131 doses selective due to the patients status, however, multinodular goiter and toxic goiter needed doses larger than graves' disease which confirmed by the study reported that, for larger multinodular goiters, doses in the range of 30 to 75 mCi may be required [15]. After I131therapy, radiation protection considerations necessary for patients to minimize radiation exposure for other people especially their family members and colleagues [16].

5. Conclusion

The majority of the patients used radioiodine therapy in this study was female. Thus, protection is more important among the cases of pregnancy, breast-feeding and for children as well as public.

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LOCAL-REFERENCE PATIENT DOSE ESTIMATION IN ROUTINE NUCLEAR MEDICINE PROCEDURES IN MAZANDARAN, IRAN

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Abstract

The study carried out a glance at nuclear medicine services in north of Iran, propose adult diagnostic reference levels (DRLs) and provide updated data on population radiation exposure resulting from diagnostic nuclear medicine examinations. The data were gathered from all departments in Mazandaran province. The seventy fifth percentile of the distribution and the average administered activity were calculated and the average effective dose per examination, annual effective dose per caput and collective effective dose to the population were estimated using dose conversion factors. Based on the gathered data, the collective effective dose was 95.628 manSv, leading to a mean effective dose of 0.03 mSv per caput. It was also observed that the myocardial perfusion was the most common procedure (50%). The seventy fifth percentile of the distribution of administered activity represents the DRL. The average administered activities and the seventy fifth percentile of the distribution of administered activity are slightly higher than DRL of other studies. Myocardial perfusion is responsible for most of the collective effective dose and it is better to review some DRL values especially myocardial perfusion scans through the participation of nuclear medicine specialists in the future.

1. INTRODUCTION

Nowadays, application of ionizing radiation in medical imaging has increased and led to the remarkable increase of collective effective dose to the inhabitants[1]. Nuclear medicine diagnostic imaging has a notable contribution in the ionizing radiation population exposure [2]. So, review of examinations in nuclear medicine departments are highly suggested in near future. The main goal of the study is estimation of radiation dose in nuclear medicine examinations toward the establishment of diagnostic reference levels (DRL) for nuclear medicine examinations.

2. METHODS

Data about various nuclear medicine procedures were collected from all active nuclear medicine departments in Mazandaran province (population = 3,155,000 person on the basis of 2011 survey) and using the standard dosimetry tables, the average effective dose per examination, collective effective dose to the inhabitants and annual effective dose per caput were estimated [3–4]. Also, the seventy fifth percentile of the distribution and the average administered activity were calculated to suggest DRLs.

3. RESULTS

The average administered activities, effective dose per examination, and the seventy fifth percentile of the distribution are presented in table 1.

TABLE 1. Average administered activities and effective dose per examination for common nuclear medicine procedure.

Procedure	Radionuclide Chemical form	Mazandaran (Average administered activity) (mci) (2003)	Mazandaran (Average administered activity) (mci) (2014)	Number of Procedures (2003)	Number of Procedures (2014)	Conversion Factor (mSv/MBq)	Effective dose per exam(mSv) (2003)	Effective dose per exam(mSv) (2014)	Mazandaran (75%) (mci) (2014)
Myocardial perfusion 2days (exercise)	Tc-99m MIBI	20	(19.05 ± 3.02)	618	4881	7.9E-03	5.57	0.15	(20.91±3.02) (21±3.26)
Myocardial perfusion 2days (rest)	Tc-99m MIBI		(17.85 ± 3.26)		4882	9.0E-03	5.94	0.16	
Myocardial perfusion 1days (exercise)	Tc-99m MIBI		(19.11 ± 3.85)		266	7.9E-03	5.59	0.15	(23.17±3.85) (14.85±3.2)
Myocardial perfusion 1days (rest)	Tc-99m MIBI		(11.43 ± 3.2)		266	9.0E-03	3.81	0.1	
Myocardium scan	Tl-201 Chloride	-	(3.05 ± 0.47)	-	200	2.2E-01	24.83	0.67	(3.64±0.47)
Bone imaging	Tc-99m MDP	20	(22.07 ± 3.4)	1861	3919	5.7E-03	4.65	0.13	(24.16±3.4)
Renal imaging	Tc-99m DTPA	15	(10.05 ± 1.18)	250	259	4.9E-03	1.82	0.05	(10.81±1.18)
Renal imaging	Tc-99m DMSA	5	(5.23 ± 1.1)	79	364	8.8E-03	1.70	0.05	(5.6±1.1)
Thyroid imaging	Tc-99m Pertechnetate	-	(6.38 ± 1.66)	1342	2505	1.3E-02	3.07	0.08	(7.58±1.66)
Lung perfusion	Tc-99m MAA	5	(5.08 ± 0.68)	63	155	1.1E-02	2.07	0.06	(5.7±0.68)
Whole body scan (after thyroid ablation)	I-131 Iodine	5	(4.46 ± 0.71)	366	360	6.1E-02	10.07	0.27	(5.12±0.71)

The highest frequency of procedure in the study were for myocardial perfusion, thyroid and bone scans. (The number of population = 3,155,000 person on the basis of 2011 survey) and the number of nuclear medicine examinations in Mazandaran was 21290. The number of nuclear medicine examinations on paediatrics was 1903 (9%). The frequency of nuclear medicine examination per 1000 population was 6.7 and the Average effective dose per examination was calculated 4.67 mSv. The collective effective dose from diagnostic nuclear medicine procedures has been estimated to be 95.628 manSv, with an annual mean effective dose of 0.03 mSv per caput. Together, myocardial perfusion, bone and thyroid scans accounted for 93.29% of examinations and 94% of collective effective dose.

4. DISCUSSIONS AND CONCLUSIONS

DRL of other studies were slightly lower than this study's results; however, in some cases, fewer countries such as Luxembourg and Spain were higher than those DRLs[5]. The estimated effective dose was indefinite due to nonuniform distribution of activity within specific organs and using biokinetic information. However, the effective dose is beneficial to estimate the risk of stochastic effects[4]. The result obtained, for the year 2014 demonstrates a tremendous shift in the type and number of examinations. The highest number of examinations was myocardial perfusion and bone scan in 2014. This result is not the same as the 2004 results. The measured data were higher than that in the 2004 results[6] but it was similar to the result of Portugal survey in 2010 and the USA study in 2008[7–8] and this may be owing to an increase in the prevalence of myocardial related-disease in this province during the last decade. In comparison with the 2004 data, the annual total number of diagnostic nuclear medicine procedures has enhanced 3 times while the population has increased approximately 9% from 2004 to 2014 years. The average effective dose increased from 4.25 mSv to 4.67 mSv. Also, the annual mean effective dose per caput, from 9.3 micro Sievert to 0.03 mSv, was obtained[6] and this per caput dose was greater than the total per caput dose from both diagnostic radiology and nuclear medicine examinations in 2004, resemble to the results indicated in the USA survey in 2007[8–9]. These increases were almost entirely because of the growth of nuclear medicine diagnostic examinations, which need higher activity to perform. The underlying frequency of nuclear medicine procedures was 6.7 examinations annually per 1000 members of population in 2014. And is lower compared to European countries ranging from 8 to 56[10] The annual mean effective dose per caput in this study such as Norway and Finland was lower than that in Australia, Switzerland and Luxembourg because of the lower number of nuclear medicine examinations in Iran[1,11]. Based on the results of our study, striking changes are noted on the trends of diagnostic nuclear medicine procedures in north of Iran during last decade. The myocardial perfusion scan because of its high frequency compared to the other types of procedures leads to the main part of this dose. Therefore it is recommended that establish the DRL of myocardium scan in the future in Iran.

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MODELING QUANUM PLATFORM FOR INTERNAL AUDITS IN MN TO COMPLY BRAZILIAN REGULATIONS

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Abstract

Audit is an ongoing review of all processes involving to ensure that each process is developed systematically and in accordance with specific regulations. The IAEA developed an audit process named QUANUM - Quality Management Audits in Nuclear Medicine, available in their website. This tool offers support to management quality audits assisting teams in the evaluation of quality management system, based on the European Community guidelines and international recommendations. In order to be better applied inside a country, national standards should also be fulfilled. A review was performed under enlightenment of the national standards and in compliance with international recommendations. Also, national requirements not addressed by international recommendations were considered. Therefore, a single model was designed to meet both requirements, national and international standards and regulations. The Internal Audit model was developed to quantify the levels of risk involved in practice, demonstrating that national regulations reach 63.63% of the international QUANUM requirements. After the modeling, a platform was obtained where 11% meet the Brazilian standards, 32% international requirements and 57% meet both. This tool systematizes and improves the quality management policy, minimizing nonconformities.

1. INTRODUCTION

Nuclear Medicine (MN) is a medical specialty that uses safe, painless and non-invasive methods to provide physiological information that other diagnostic exams cannot. This technique uses non-sealed sources of radionuclides, named radiopharmaceuticals, administered to the patients orally, by inhalation or subcutaneously, presenting a specific distribution for each organ or cellular tissue, functions of the labeled chemical [1]. Although the vast majority of procedures in Nuclear Medicine have a diagnostic purpose, therapeutic procedures have also been performed for many years. In these cases, the objective is the deleterious action of the ionizing radiation emitted by the radiopharmaceuticals in a tissue or organ of interest. As examples of these procedures we can highlight the use of iodine-131 in the form of sodium iodide for thyroid disorders treatments, and Samarium-157 in the form of Hydroxyapatite used in treatments for bone pain [1].

As a multidisciplinary field, Nuclear Medicine involves several processes, ranging from molecules labeling in radiopharmaceutical preparation, evaluation of the performance of diagnostic equipment, patient dosimetry, radiological protection, clinical analysis, among others. Thus, the effective control of the parameters that ensure the quality of care for the population for diagnostic and therapeutic applications should be performed periodically, in order to maintain a satisfactory level [1].

The Quality Control (QC) of the equipment is part of a larger Quality Assurance (QA) program. Not least, the training of the professionals involved should also be evaluated during the implementation of the QA. This involves all efforts to ensure better results for the radiation detection process that will result in the best possible image, with a safety and optimized technique [1]. The International Atomic Energy Agency (IAEA) has developed a tool named *QUANUM-Quality Management Audits in Nuclear Medicine Practices*, which supports quality management audits in nuclear medicine practices, assisting audit teams in this evaluation [2].

Brazil has two regulatory bodies for NM: National Nuclear Energy Commission (CNEN) and National Sanitary Surveillance Agency (ANVISA). CNEN establishes the CNEN-NN-3.01 [3] standard to lay down basic

requirements for radioprotection and CNEN-NN-3.05 [4] to specify requirements for the licensing and control NM facilities [5]. Furthermore, ANVISA establishes sanitary requirements in Resolutions of the Collegiate Board, or RDC. For specific NM control and regulation ANVISA published RDC-38 on June 4, 2008, providing design and operation conditions for *in vivo* Nuclear Medicine services [5].

However, none of the regulatory agencies has a published methodology to verify compliance with specifically applicable requirements or conformities assessed by documentation or on-site inspections. This self-assessment, or internal audit, is required to comply with the QA process ensuring satisfactory performance. For this internal audit to be adequate, it is necessary to establish a methodology through an evaluation process containing all information of the steps that constitute a QA system [6]. To address this issue, the QUANUM tool based on European Community guidelines was adapted to Brazilian standards and regulations to be applied in nuclear medicine services into the country [2].

2. METHODOLOGY

Brazilian standards and regulations and international recommendations was analyzed and compared to requirement introduced by QUANUM tool. Within the worksheet, a new tab was created to identify Brazilian legislation items. Therefore, a single model was designed to meet both requirements, national and international standards and regulations. The process should contain the quality assessment of all components related to the practice, including professional education and training needs to be continuously evaluated [7,8].

In the developed model a "risk level" has to be evaluated related with the level of compliance; risks was defined as A, B and C. Level "A" was classified as *high risk*, where an immediate solution for noncompliance should be provided. Level "B" was considered *moderate risk* where nonconformity has to be solved in a period ranging from 1 to 3 months. Level "C" was considered a *low risk* and the nonconformity can be solved within 6 months or until the next internal audit.

Then, a National Internal Audit model was developed to be capable to evaluate the real level of competence of a NM service, taking into account the management, operational procedures, installation, equipment, human resources, impact on operational practice and records. This audit system should be able to evaluate from the patient's entry into the clinic until the exit of the medical report.

3. RESULTS

Currently in Brazil there are 460 nuclear medicine services, of which only 39 are registered in the IAEA NumDAB system (IAEA Nuclear Medicine Database). The registration of a service in the system is voluntary but is important to inform the level of assistance and the real competence is in place. Table 1 shows actual status of this practice in Brazil.

TABLE 1: STATUS OF THE NUCLEAR MEDICINE PRACTICE IN BRAZIL, 2016.

Number of installed equipment	Planar and Single Photon Emission Tomography –SPECT	930
	Single Photon Emission Tomography/Computed Tomography –SPECT/CT	10
	Positron Emission Tomography/Computed Tomography – PET/CT	130
	Cyclotron for radionuclide production	18
Therapeutic rooms	For Radioiodine Therapy	100
Radionuclides in use	^{99m}Tc, ^{131}I, ^{123}I, ^{18}F, ^{11}C, ^{67}Ga, ^{68}Ge, ^{111}In, ^{153}Sm, ^{177}Lu, ^{90}Y, ^{201}Tl, ^{223}Ra	

The number of Nuclear Medicine services in the country is growing fast and the complexity of procedures and controls makes necessary to better supervise and guarantee the safety of these installations.

Analyzing national regulations, the quality control tests (QC) of the equipment are stablished without any minimum performance parameter. Therefore, it is necessary to elaborate minimum performance values to be implemented, based on international and manufacturer recommendations for good equipment performance and

tests reliability [3]. The Regulatory Authorities also do not establish minimum requirements for professional training as doctors, pharmacists, physicists, nursing technicians, radiology technicians, among others.

The adapted QUANUM tool is demonstrated in figure 1, where it could be observed international (blue) and nationals (red) requirements to each evaluated item.

QUALITY MANAGEMENT AUDITS IN NUCLEAR MEDICINE

CHECKLIST 5 *PATIENT RADIATION PROTECTION*	CHECKLIST SUMMARY	N.	APPLICABLE	TOTAL SCORE	% SCORING	NC	

Nº	COMPONENT	CONFORMANCE LEVEL	RISK LEVEL	COMMENTS/PLANNED ACTION	DATE ACHIEVED	EXAMPLE OF RESULT/ TYPE OF EVIDENCE	INTERNATIONAL REFERENCES	BRAZILIAN REFERENCES
5.1	Are there standard operating procedures (SOPs) available to ensure correct identification of the patient prior to administration of the radiopharmaceutical?					Check the procedure/ Observation on site.	SRS No.40, par 3.2.1, 5.2,5.3 BSS, par. 3.156	CNEN NN 3.05 - Cap.I,Seç.V,Art.13,IV.c
5.2	Are there SOPs and appropriate signage for alerting female patients of child bearing age to report any potential pregnancy or breast-feeding?					Check the procedure/ Observation on site.	SRS No.40,par 2.3.3,5.3.1,5.3.2 BSS, 3.151,3.165,3.174-3.176	CNEN NN 3.05 - Cap.III, Seç.VI,Art.45 RDC 38 - 5.2.3
5.3	Are written instructions available and verbal instructions given to patients before and after administration of radiopharmaceuticals?					Observation on site/ copy of the instructions.	SRS No. 40, par 3.2.1,5.2,5.3 BSS, par. 3.150 - 3.152	CNEN NN 3.05 - Cap.I, Seç.III, RDC 38 - 5.3, a, b.

FIG.1. QUANUM tool with international and national data.

A quantitative comparison between the requirements evaluated in international recommendations and standards with those requested in the national ones were performed and the results are shown in figure 2. It can be observed that national regulations and standards meet 63.63% of the international QUANUM requirements, figure 2.

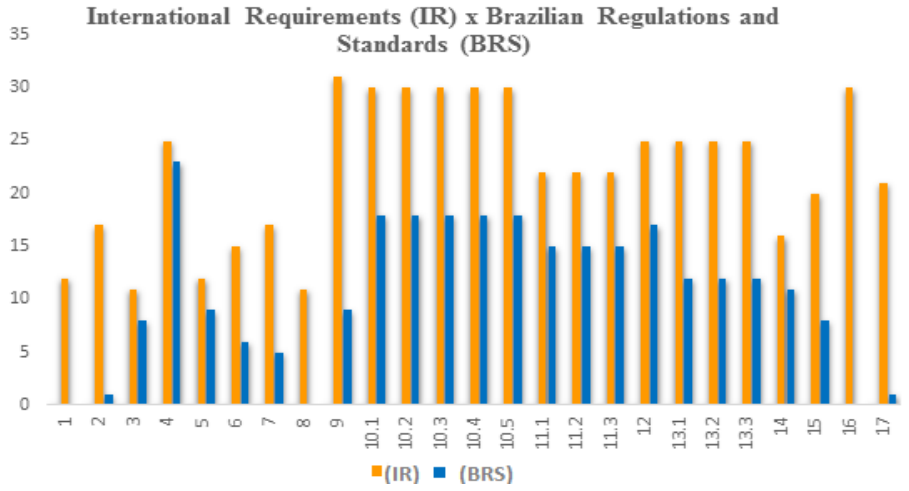

FIG. 2. Comparison of national regulations and standards with international requirements.

After the modeling QUANUM platform to Brazilian normative requirements, the obtained platform has 11% requirements to meet Brazilian standards, 32% to international requirements and 57% for both, as we showed in figure 3. Then, a unified model was elaborated, and it could be applied to the SMN of Brazil¸ also meeting international requirements.

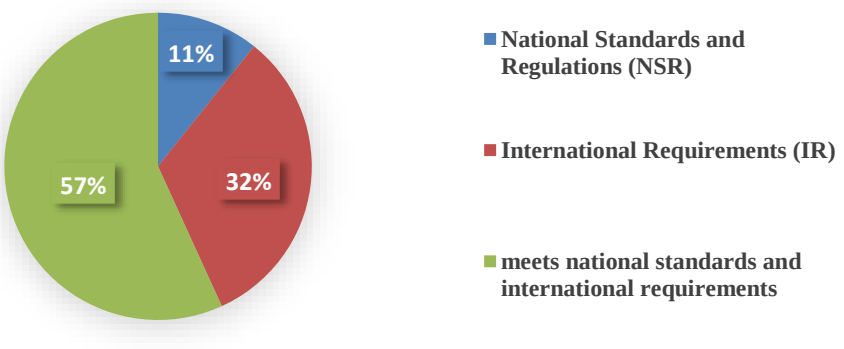

FIG. 3. QUANUM model adapted to Brazilian standards.

4. CONCLUSION

An easy tool to evaluate quality of the health system applied to Nuclear Medicine Services in Brazil was developed based in QUANUM process proposed by IAEA. The Internal Audit (IA) model will attend national regulations and can help to quantify risk levels concerned to the process as a whole. This tool systematizes and improves the quality management policy in this field and, at last, is be able to attend Regulatory Audit (RA) minimizing non-conformities.

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THREE-MONTH EYE RADIATION EXPOSURE MONITORING OF NUCLEAR MEDICINE DEPARTMENT PERSONNEL.

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Abstract

New EURATOM Council Directive 2013/59 modifies the annual occupational dose limit for the eye lens to 20 mSv from the previous value of 150 mSv. The study's aim is to investigate the level of eye lens doses to the personnel in a typical nuclear medicine department. Over a three-month period the eye radiation exposure of 10 staff members (2 physicians, 3 nurses and 5 technicians) handling radiopharmaceuticals was measured using thermoluminescent dosimeters. Detailed data on the radioactivity of radiopharmaceuticals used by employees were analysed.
Eye dose results were compared with corresponding hand and whole body doses obtained for these staff members. The highest eye doses were: 0.72 mSv, which was for a technician, 0.41 and 0.27 mSv for 2 nurses. The highest dose per unit of radioactivity was 0.027 μSv/MBq, measured for a physician administering iodine I131. Correlations of 0.52 between eye and hand doses and 0.61 between eye and whole body doses were obtained. Eye doses were found to be well within the new proposed annual limit. Doses to the whole body of staff preparing and administering radiopharmaceuticals may be a relative indicator of eye doses in the monitored Nuclear Medicine Department.

1. INTRODUCTION

Working with open sources of ionizing radiation requires permanent dosimetric control. The aim of this study is to investigate the level of eye doses and the correlation between other doses, such as hand doses and whole body doses, for staff in a classical nuclear medicine department. In the paper the eye lens exposure in workers handling I-131 and Tc-99m isotopes was analysed. The results obtained in the study might be a useful tool for the analysis of staff exposure and to improve conditions of radiological protection in other nuclear medicine centres.

2. METHODS

One of the purposes of this paper was to compare values of personal dose equivalents for the eye lens, whole body and fingers measured by different kinds of individual dosimeters. EYE-D ™ dosimeters with a high-sensitivity MCP-N detector were used for 3 months at the Nuclear Medicine Department of the Pomeranian Medical University in Szczecin, and then were provided to and measured by the Institute of Nuclear Physics, Polish

Academy of Sciences (IFJ PAN) in Cracow. Doses which were measured in terms of personal dose equivalent: Hp(3) for the eye lens, Hp(0.07) for hands, and Hp(10) for the whole body. Over a three-month period the radiation exposure of 10 staff members, who were divided into three occupational groups: technicians-T, nurses-N and physicians-P, was measured. The paper examines the two most commonly used isotopes in classical nuclear medicine: I-131 for therapy and Tc-99m for diagnostic imaging. Of the technicians (5 workers), four of them were involved in the labelling procedures in a hot lab, using only Tc99m, in the weekly rotation, and patient imaging. One technician was also working with I-131. The nurses, a group of three, were mostly working with Tc99m during the labelling procedures in a hot lab and giving patients radiopharmaceuticals by injections or orally. One of them was also working with I-131. Physicians, two in number, were exposed to I-131 during administration of I-131 therapeutic capsules to the patients. Radioisotope administration of Tc-99m was performed using lead-shielded vials and tungsten-shielded syringes. Hp(10) as well as Hp(0.07) were measured during the whole quarter of 2016 (over three months: from 1 Oct to 31 Dec). Hp(3) equivalent doses were measured during two periods: from 1 Nov to 31 Dec, and separately, from 1 to 31 Jan 2017. The two dose records of Hp(3) for workers have been added up in the analytical part of this work.

In addition, it was investigated whether there was a correlation between doses for the workers' eyes and doses for the whole body and hands. For this purpose the Spearman test was performed, taking into account the readings of two-month eye doses of Hp (3) and quarterly doses for the whole body Hp (10) and hands Hp (0.07) for the analyzed group of workers.

3. RESULTS

The values of personal dose equivalents and total activities of Tc-99m and I-131 for each member of medical staff are presented in Table 1.

TABLE 1. PERSONAL DOSE EQUIVALENTS IN MEDICAL STAFF AND TOTAL ACITIVITY OF ISOTOPES USED BY STAFF OVER 3 MONTHS.

Medical staff	Hp(0.07) mSv	Hp(10) mSv	Hp(3) mSv	Activity [GBq]	Isotope
Physician 1	0.1	0	0.1+0.1	11.42	I-131
Physician 2	0.73	0	0.1+0.1	14.6	I-131
Technician 1	1.23	0	0.1+0.1	58.9	I-131
Technician 2	8.08	0.12	0.1+0.1	136.5	Tc-99m
Technician 3	26.05	0	0.62+0.1	151.67	Tc-99m
Technician 4	2.76	0.25	0.11+0.1	247.8	Tc-99m
Technician 5	0.1	0.97	0.26+0.1	423.56	Tc-99m
Nurse 1	7.22	0.27	0.16+0.13	471.59	Tc-99m
Nurse 2	4.67	0.26	0.15+0.11	448.01	Tc-99m
Nurse 3	19.26	0.56	0.28+0.13	645.31	Tc-99m

Table 2 shows the average doses for exposed employees who were divided into three occupational groups.

TABLE 2. AVERAGE DOSE RANGE WITH STANDARD DEVIATIONS FOR EACH OCCUPATIONAL GROUP

Occupational group	Average Hp(0.07) ± SD (mSv)	Average Hp(10) ± SD (mSv)	Average Hp(3) ± SD (mSv)
Physicians- P	0.4±0.2	0.0± 0.1	0.2± 0.1
Technicians- T	7.7±2.2	0.3± 0.1	0.3± 0.2
Nurses- N	10.4±2.6	0.4± 0.1	0.3± 0.1

Figure 1 shows the ratio of personal dose equivalents: Hp(10) to Hp(3) and Hp(0,07) to Hp(3) for each occupational group.

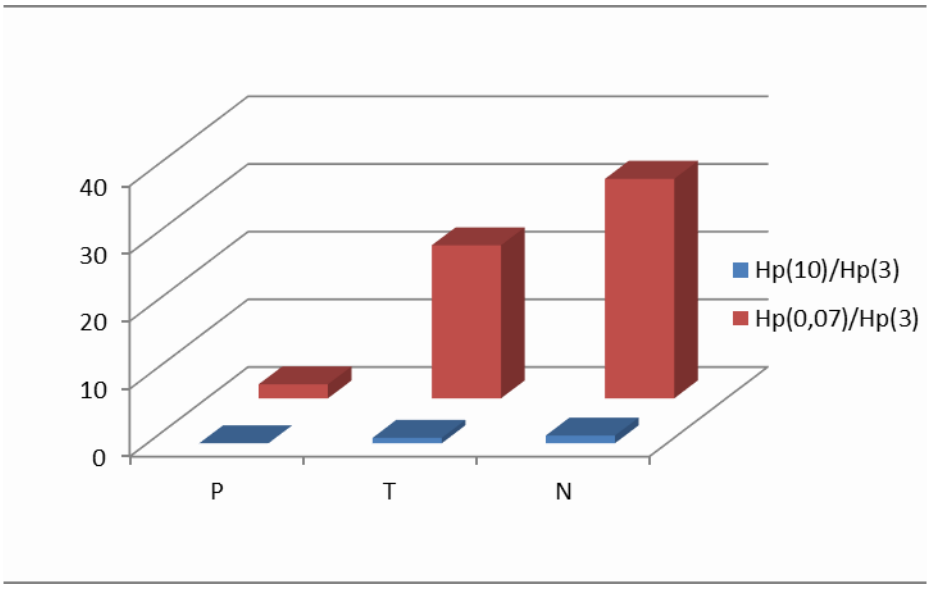

Fig 1. Ratio of personal dose equivalents measured over 3 months for three different occupational groups: p- physicians, t- technicians, n- nurses

Spearman's correlation coefficient of 0.52 between eye and hand doses and 0.61 between eye and whole body doses were obtained for exposed staff.

4. DISCUSSION

In the department of nuclear medicine, where over 95% of the procedures are performed by technicians and nurses using Tc99m, only one technician and two physicians were exposed to I-131. The highest activity of Tc-99m was 645.31 GBq for one of the nurses, who received the highest doses of Hp(0.07) and Hp(3) in this professional group. Even one technician who handled the highest activity Tc99m did not receive the highest dose equivalents Hp(0.07) and Hp(3). These values may be explained by the style of working and the training (Table 1). Kopeć et al. (2011) showed that ratios of Hp(3) and Hp(10) ranged between 0.7 and 1.1. A broader range was observed by Dabin et al. (2016), where the ratios of Hp(3)/Hp(10) were from 0.3 to 2.3. In our work the values were between 0.9 and 1.3, i.e. much closer to those presented by Kopeć et al. In Table 2 the average dose range with standard deviations for each occupational group is shown. The level of the eye lens dose in each occupational group is very similar, and there is a weak correlation between Hp(3) and Hp(10). Big differences between values of Hp(0.07) for different occupational groups were observed, and hence the ratio of Hp(0.07)/Hp(3) was from 2.1 to 32.4 (Figure 1).

5. CONCLUSIONS

In the paper the individual doses in terms of Hp(0.07), Hp(3) and Hp(10) for medical staff, i.e. physicians, technicians and nurses in a nuclear medicine department were evaluated. Practically, the exposures of the eyes and the whole body were similar, while the hand exposure was the highest in the nuclear medicine department, especially for nurses involved in preparing and injecting radiopharmaceuticals and for technicians involved in preparing radiopharmaceuticals and patient imaging. Eye doses were found to be well within the new proposed annual limit. Doses to the whole body of staff may be an indicator of their eye doses in the monitored Nuclear Medicine Department.

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ESTIMATION OF RADIATION DOSES RECEIVED BY NUCLEAR MEDICINE STAFF FROM ^{18}F-FDG DISPENSING

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Abstract

The purpose of this study is to estimate the radiation doses to whole body and extremities (finger) per cycle received by the PET/CT radiochemists & radio-pharmacists during 18F-FDG automated dispensing process in a PET/CT facility at Shaukat Khanum Memorial Cancer Hospital & Research Centre, Lahore Pakistan. The whole-body radiation doses were measured and evaluated using EPDs (Electronic Pocket Dosimeters) while the doses received by fingers were measured using TLD Ring dosimeters. All these doses were evaluated for the period of 6 months. In 68 FDG dispensing cycles, the average whole-body dose per cycle received by the radiochemist / radiopharmacist was 2.54±0.10 μSv while the average finger dose per cycle received was 4.82±1.10 μSv. The average time of contact with 18F-FDG per cycle was 5.30±1.00 minutes. The accumulated radiation doses shows that in comparison with the ICRP dose limits, each nuclear medicine staff can work comfortably with 18F-FDG dispensing facility without exceeding the occupational dose limits. This study shows that personnel involved in 18F-FDG dispensing in our facility received low levels of radiation doses and the working environment is safe from radiation exposures.

1. INTRODUCTION

Positron Emission Tomography /Computed Tomography (PET/CT) has an important role in tumor diagnostic imaging. Fluorine-18 Fluorodeoxyglucose (FDG) is a radiopharmaceutical used in the medical imaging modality (PET). 18F-FDG is a positron emitter radiopharmaceutical and is used clinically in PET imaging. This radiopharmaceutical has been successfully used diagnosing tumors, metastasis, tumor staging and early detection of recurrent malignant diseases [1]. PET/CT also has a very wider role in treatment planning and to monitor therapies. F18 has 0.511 MeV annihilated photons due to positron decay and has higher energy than other diagnostic imaging radiations. The exposure rate constant for F18 source in air (1.49 x 10–4 mSv. m2/MBq.h) is approximately 7 times greater than other commonly used radio tracers in nuclear medicine procedures [2].

The PET/CT facility at Shaukat Khanum Memorial Cancer Hospital & Research Centre Lahore started functioning in February 2010. In our facility, the numbers of PET/CT scans are increasing every year. Due to increase in PET/CT scans and the higher energy of F18 has raised the issues of radiation exposure to radiochemists and radiopharmacist while dispensing and preparing 18F-FDG [3,4,5].

The purpose of this study is to estimate the radiation doses to whole body and extremities (finger) per cycle received by the PET/CT radiochemists & radio-pharmacists during 18F-FDG automated dispensing process in a PET/CT facility at Shaukat Khanum Memorial Cancer Hospital & Research Centre, Lahore Pakistan, and to evaluate the results against occupational limits recommended by International Commission on Radiological Protection (ICRP).

2. MATERIALS AND METHODS

MGP Model DMC 2000GN Electronic Personal Dosimeter) was used to measure the whole body radiation doses. This EPD has measuring gamma energy range of 40keV to 6MeV and dose display range from 1μSv to 10Sv. The dose rate linearity dependence of this EPD is <+/- 10% up to 1 Sv/h. These EPDs were

calibrated with a Cs-137 radioisotope in the Secondary Standard Dosimetry Laboratory (SSDL) of Pakistan Institute of Nuclear Science and Technology (PINSTECH), Islamabad Pakistan.

The thermoluminescence (TLD) ring dosimeters were used to measure the finger doses. These rings are monthly processed and evaluated at Radiation Dosimetry Group, Health Physics Division (HPD), PINSTECH, Islamabad Pakistan.

Radiochemists / radiopharmacists belong to the group of radiation workers who are potentially exposed to radiation doses. These radiochemists and radiopharmacists work in rotation to perform 18F-FDG dispensing and preparation duties. In our facility, routinely around 32±1 GBq per cycle is dispensed and prepared using Theodorico Robotic Arm. All these workers were guided about the method for measuring and recording their radiation doses received in a cycle.

2.1. Whole-body dose measurement

Each individual radiation worker was provided with one EPD to use throughout 18F-FDG study. Workers were advised to place the EPDs on the upper left pocket of lab coat. The values of the radiation doses received were recorded directly from the dosimeter at the end of each cycle of 18F-FDG dispensing and preparation. These doses were corrected by the calibration factor obtained from the SSDL PINSTECH, Islamabad Pakistan. The average value of the radiation doses received by each individual was calculated. The time spent by each cycle of dispensing and preparation was also recorded. This study was carried out for the period of 6 months.

2.2 Finger dose measurement

TLD rings were worn on the index finger of working hand. These rings were used during the procedures only and otherwise were kept in lead shielded box. These TLD rings were evaluated at HPD PINSTECH, Islamabad Pakistan. The average value of the radiation doses received was calculated. The values of the doses received by fingers over the period of 6 months are used in this study.

3. RESULTS

A total of 68 ^{18}F-FDG dispensing and preparation cycles were studied. The 18F-FDG activity per cycle was 32±1 GBq. The average time of contact with ^{18}F-FDG during dispensing & preparation was 5.30±1.00 minutes per cycle. The average whole body effective doses (μSv) and finger doses per study received by the workers are reported in Table 1 & Table 2 respectively.

TABLE 1

Radiation Workers	**Number of 18F-FDG dispensing Cycles**	**Effective Whole Body Dose (μSv) / Cycle (study)**
Radiochemist 1	38	2.36 ± 0.10
Radiochemist2	16	3.09 ± 0.11
Radiopharmacist	14	2.16 ± 0.09
	Average:	**2.54 ± 0.10**

TABLE 2

Radiation Workers	**Number of 18F-FDG dispensing Cycles**	**Finger Dose (μSv) / Cycle (study)**
Radiochemist 1	38	4.50 ± 1.10
Radiochemist2	16	5.13 ± 1.13
Radiopharmacist	14	4.83 ± 1.07
	Average:	4.82± 1.10

4. DISCUSSION

Table 1 shows that, during 68 FDG dispensing cycles, the average effective whole-body dose per cycle received by the radiochemist / radiopharmacist was 2.54±0.10 μSv while the average finger dose per cycle received was 4.82±1.10 μSv. The average time of contact with ^{18}F-FDG per cycle was 5.30±1.00 minutes.

If each of our radiation workers works with maximum capacity e.g. 250 ^{18}F-FDG dispensing cycles during one year, the maximum effective whole body radiation doses would be around 1mSv and the maximum finger doses would be about 1.5mSv

The accumulative radiation doses show that in comparison with the International Commission on Radiation Protection (ICRP) dose limits [6], each nuclear medicine staff can work comfortably with ^{18}F-FDG dispensing facility without exceeding the occupational dose limits.

5. CONCLUSION

This study shows that personnel involved in 18F-FDG dispensing in our facility received far low levels of radiation doses as described by ICRP and the working environment is safe from radiation exposures.

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QUALITY MANAGEMENT AUDIT IN NUCLEAR MEDICINE PRACTICES AT OCEAN ROAD CANCER INSTITUTE, TANZANIA

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Authors' disclosures of competing conflicts of interest and contributions are found at the end of the article.

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Abstract

Basic Nuclear Medicine services at Ocean Road Cancer Institute were established in 1994 and since then, positive and progressive improvement in service delivery has been pursued. The department is performing diagnostic and therapeutic procedures and it was an opportune time for a formal auditing of the department to assess the quality of service delivery, radiation safety and protection. The objective was to evaluate the quality of service delivery, radiation safety and protection at the Nuclear Medicine Department. Methods: An IAEA multidisciplinary internal quality audit team was formed in March 2013. After initial assessment, the team gave a deadline of three months for the department to prepare all standard operating procedure (SOP). The IAEA multidisciplinary external audit team visited Ocean Road Cancer Institute in May 2014. The Team checked the use of SOPs in the department, guided by the IAEA manual. The main findings were; the SOPs were available but partially used, patient protection and radiation safety regulations were above 60%, and overall departmental performance was 65%. Radiopharmacy performance ranged between 30%-50%. In conclusion: Nuclear Medicine services, radiation safety and protection were acceptable but there was a need for improvement.

Background

Nuclear medicine (NM) is a medical specialty that uses radiopharmaceuticals to evaluate molecular, metabolic, physiologic and pathologic conditions of the body for the purposes of diagnosis, therapy and research [1]. The role of NM in combating non-communicable diseases (NCD) especially cardiac and oncologic diseases is well established worldwide [2].

Despite this knowledge, the NM services are scarce in low- and middle-income countries (LMICs) [3]. The cost of establishing NM is a hindrance to many countries in Africa, hence almost all countries in Sub-Saharan Africa have acquired their first NM facility through an International Atomic Energy Agency (IAEA) technical cooperation programme [4].

IAEA supported the establishment of the NM service in Tanzania, and the service was established in 1994 at Ocean Road Cancer Institute (ORCI). IAEA supported training of NM staffs, acquisition of several NM equipment as well as IAEA consultancy/supervision so as to make sure the service is sustainable.

Together with maintaining service sustainability, ensuring radiation safety among workers, patients and the public is very important [5]. Taking this into consideration, IAEA developed a Quality Audit In Nuclear Medicine Practices (QUANUM) tool to guide evaluation of NM services including radiation regulations, safety compliance and radiation protection, through internal and external audits [6].

An external audit by IAEA to assess the quality of NM practices at ORCI was held on 19th - 23rd May, 2014 being preceded by an internal audit in March 2013. The aim of the audit was to achieve quality improvement of NM practice at ORCI.

Methods:

The ORCI NM Department is the oldest in the country being established in 1994. The department has a SPECT single head gamma camera (Siemen), dual head SPECT gamma camera (Mediso), lamina flow cabinet, 2 dose calibrators, 2 gamma counters and equipment necessary for cardiac imaging. Both diagnostic and therapeutic procedures are offered by the department. Radiopharmaceuticals are imported from Amersham South Africa.

In March 2013, a NM internal (hospital) quality audit team was formed following instructions from IAEA. The internal audit team consisted of a member from each department of the Institute that met monthly and checked availability of various items and verified activities in the NM department as indicated in the IAEA QUANUM tool.

Initially, this exercise was difficult to NM staff, since the documentation system was not systematic and all procedures were routinely done, and there were no formal instructions for the procedures performed in the department.

After the initial audit by the internal team, a deadline of three months for the department to prepare all standard operating procedure (SOP) for each activity done in the department was given. It took six months for the internal audit team to be satisfied with the improvements, and request for an expert audit from the IAEA's external audit team.

The IAEA audit team had five (5) members; a physician, a physicist, a technologist, an IAEA technical officer and a radiopharmacist. The audit was done for five consecutive days from 19th - 23rd May, 2014. The team checked the physical availability of every item including SOP's, the use of SOPs and documentation in all sections of the department as guided by the QUANUM tool.

All aspects of NM department operations were checked and this comprised administration and human resource development, radiation regulations, safety compliance and radiation protection, quality control of imaging equipment, data handling, radiopharmacy and assessment of diagnostic and clinical services. This tool

was completed by both the internal and the external teams during the two audits.

The tool gave the following choices for each item to be checked: 1) not applicable, absent or inappropriate 2) planned or approximate 3) partially conform or partially implemented 4) largely conform or largely implemented 5) fully conform or fully implemented. These choices were allocated scores from 0-4. The choices number 1 remained as not applicable with color code white, number 2-3 where grouped as non-conformance coded with color red, and 4-5 were conformances with green color. The overall data was illustrated in a radar plot.

Results

Overall the department had 35 critical, 46 major and 21 minor non conformances. The critical non conformances were the one directly affecting patients' safety or represents risks to staff or environment.

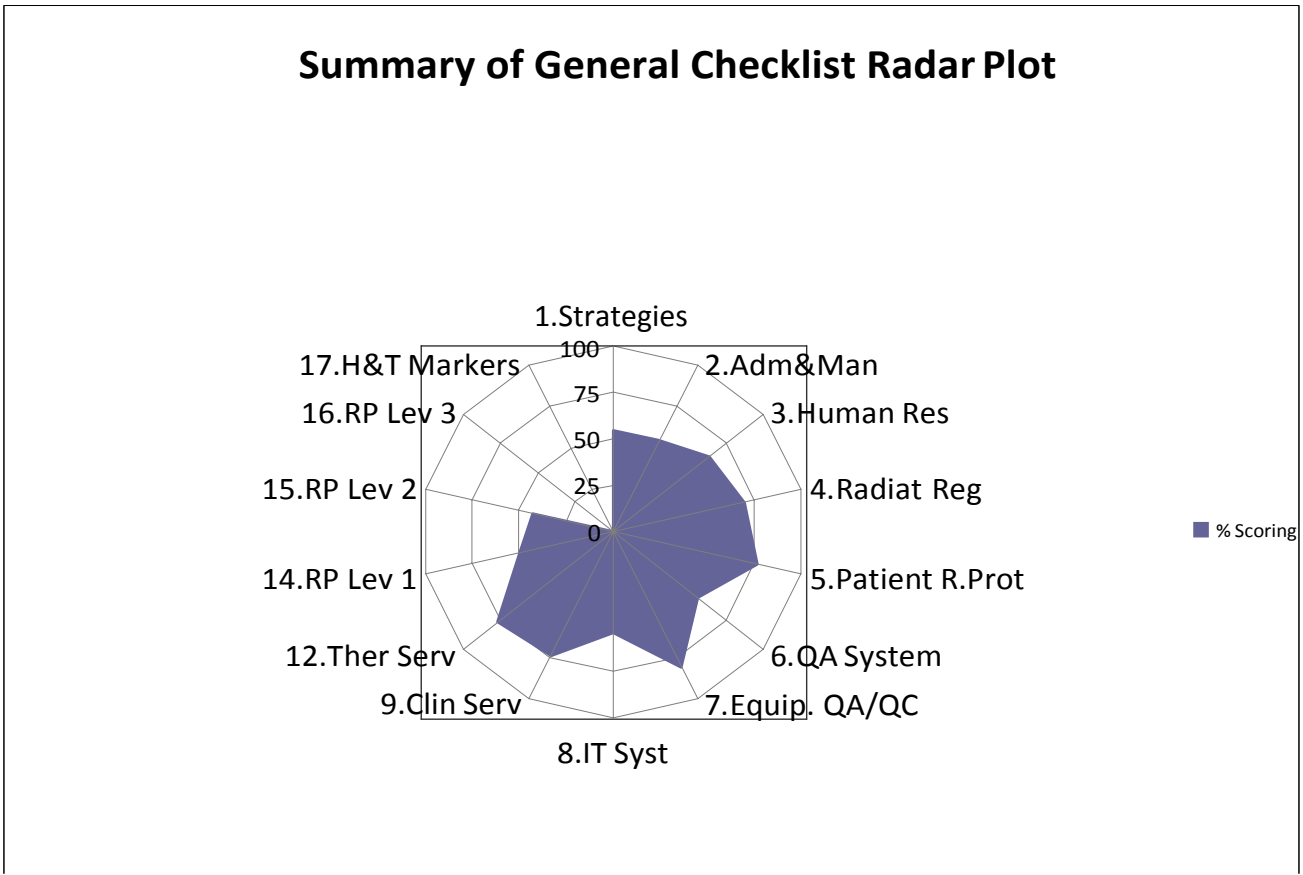
The difference in audits between internal team gave the department the overall score of 64.7% and external team scored 61.1% as seen in table 1.

Table 1; comparison of internal and external audits

Summary of Checklist	Internal audit score	External audit score
N. Questions	233	233
Applicables	195	183
NA	68	80
Total score	505	447
% scoring	64.7	61.1
N. of NC	94	88

The radar plot from internal audit showed radiation regulation was above 60% and patients' radiation protection to be about 75%. The lowest score was on NM strategies and was followed by quality control for imaging equipments, computer system and data handling (Figure1)

Figure 1. Internal audit team results

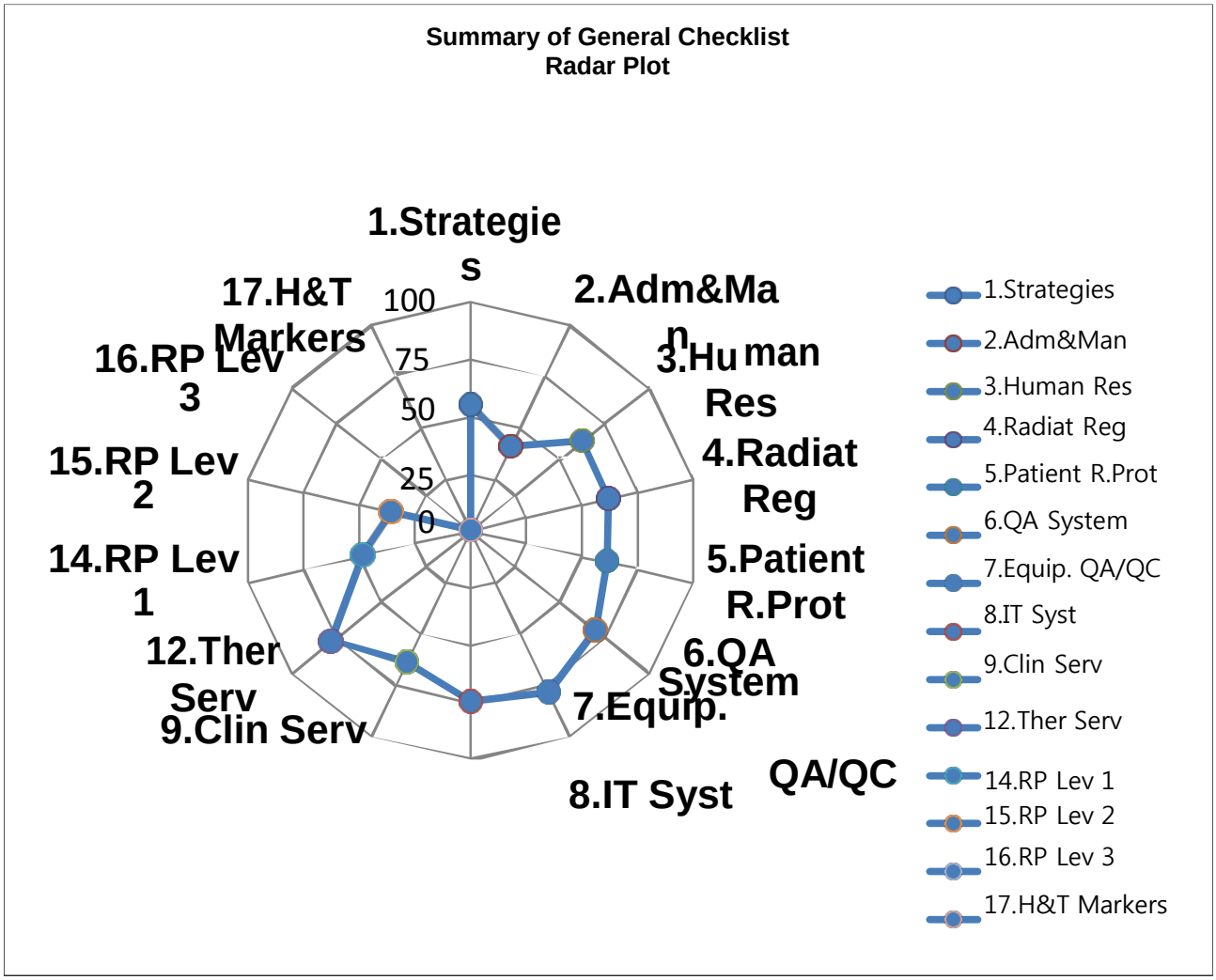


Key:

1. Adm&Man – administration and management
2. Human Res – human resource development
3. Radiat Reg – radiation regulation
4. Patient R. prot – patient radiation protection
5. QA System – evaluation and assurance of quality systems
6. Equip. QA/QC – quality control for imaging equipments
7. IT Syst – computer system and data handling
8. Clin Serv – general clinical services
9. Ther Serv – general radionuclide therapy
10. RP Lev 1 – radiopharmacy operational level I
11. RP Lev 2 – radiopharmacy operational level II
12. RP Lev 3 – radiopharmacy operational level III
13. H&T Markers – hormones and tumor markers

Following the same tool, the external audit team scored quality control for imaging equipment, computer system and data handling with the highest score which was above 75% and the lowest was radiopharmacy, administration and management (Figure 2)

Figure 2: IAEA audit results



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Discussion

Overall the NM practice at ORCI scored above 60% by both internal and external audit team. This is very encouraging taking into account that Tanzania is a LMIC with a double burden of communicable and non-communicable diseases [7] Tanzania emphasizes on quality health care focusing on professionalism, whereby the contribution of NM procedures in patient management is well understood even though not fully utilized due to financial constraints.

The difference between the two teams was very small (less than 3%), this is because of the uniformity and reproducibility of the QUANUM tool which makes it a very powerful tool in assessing quality of NM practice provided by different departments across the world especially in LMICs [8].

The two teams showed that radiation regulations and patients' radiation protection to be above 50%. This sends a strong message to NM departments elsewhere especially in LMICs, that despite challenges faced on sustaining the NM services, all radiation protection rules should be fully observed [9]. ORCI is working to meet all the challenges so as to fully abide to the radiation protection rules.

NM practice is new to most LMICs especially in sub Saharan region [3], and this could account for lack of strategies, administration and management policies especially on sustenance.

A high index (>75%) in the rightful use of radioactive Iodine radionuclide therapy for thyroid diseases has been recorded in the audit despite the challenges of handling radioactive iodine [10]. This proves that with good planning and management, it is possible to maintain high level of radiation protection in the Nuclear medicine departments even in LMICs.

Conclusion: Critical adherence to the SOPs and appropriate documentation of processes and procedures are needed so as to reduce the number of non-conformances and hence improve the radiation protection and quality of NM practice at ORCI in Tanzania.

Authors' contributions

LS: designed the study, collected the data, made the analysis, wrote the manuscript

KM: supervised study designing, data collection, data analysis and manuscript writing

Authors' disclosures of potential conflicts of interest

All authors declare that there are no relationships to disclose.

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WHOLE-BODY ^{18}F-FDG PET-CT WITH CT LOW DOSE SCANNING: RADIATION LEVELS FOR ONCOLOGIC DIAGNOSIS

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Abstract

The effective dose from low dose CT was evaluated using lithium fluoride thermoluminescent detectors activated with magnesium and titanium (LiF: Mg, Ti - TLD-100) Rod, inserted in anthropomorphic Alderson Randon® male phantom points corresponding the most radiosensitive organs and greater likelihood of exposure. After thermoluminescent detectors insertion, the phantoms were subjected to the same protocol of image acquisition which the patients were submitted, and the irradiation field of the CT skull base at the root of the thigh. As for the contribution determiner effective dose of the PET, we used estimates according to the biokinetic model proposed by ICRP 106 for the radiopharmaceutical 18F-FDG. In diagnostic PET scans, the activity radioactive injected into the patient is calculated based on their body mass. This work was considered a factor of 3.7 MBq / kg of patient. The average effective dose from the examination of PET-CT with low dose CT was (8,51 ± 2,21) mSv in male anthropomorphic phantom. The effective dose from low dose CT scanning corresponds to approximately 43% of the effective dose in a PET-CT.

1. INTRODUCTION

The PET-CT equipment are constituted by coupling two scanners, one a Positron Emission Tomography (PET) and a helical Computer Tomography (CT) diagnostic quality allowing the fusion of metabolic PET images with anatomical CT images. This diagnostic test associate the high PET metabolic sensitivity and the high CT spatial resolution, to be performed anatomical correlation that before was impossible. This image generated characteristic on the PET-CT enables early detection and precise lesion location.

In Brazil are estimated for the year 2016/2017 approximately 596 000 new cancer cases. Many of these cases can be diagnosed and staged by PET-CT technology. Therefore, it is important to know the patients radiation levels are subjected.

By having diagnostic technique coupled the patients subjected to higher radiation levels (Huang, 2009). However, the reliability and the early diagnosis possibility in clinical indication cases outweigh the detriment caused by the high energy deposition in the patient.

The radiation highest level is due to the fact that the patient also receives the radiation levels from the radiopharmaceutical administered and the radiation emitted by computerized tomography. Another important factor related to the patients radiation levels undergoing in this diagnostic technique is the fact that protocols be adapted according to the CT type equipment used. Thus, the modification of the protocols technical parameters is crucial to different image acquisition technique.

For existing different protocols and equipment, it is important to perform dosimetric assessment in patients undergoing diagnostic ionizing radiation. In addition, knowledge patients doses is part of the new culture radiological protection, being a requirement in many countries (ICRP, 2007).

The organ absorbed dose and effective dose study in patients undergoing PET-CT for cancer diagnosis and staging allow the optimization of diagnostic radiology procedures and demonstrate the radiological protection

principles application (ICRP, 2006). This way is possible to obtain diagnostic quality images with minimal patient exposure.

2. METODOLOGHY

To absorbed dose and effective dose study in patients undergoing PET-CT for oncological diagnostic with low dose CT scanning, we used the equipment PET-CT Discovery 690 (D-690) manufacturer from General Electric (GE).

The absorbed and effective doses evaluation from CT, were done using the anthropomorphic phantom Alderson Rando® in the male version. This phantom consists of a human skeleton wrapped in a polymer material with equivalent woven characteristics.

The anthropomorphic phantom male version has the standard man dimensions with 1.75 m and 73.50 kg (OLIVEIRA, 2012).

To absorbed doses evaluate, dosimeters were positioned within this phantom at interest points. The dosimeter made of lithium fluoride doped with magnesium and titanium (TLD-100 Mg, Ti LiF) in Rod version was manufactured by Harshaw Chemical Company and was previously selected and calibrated. These dosimeters have 1 mm in diameter and 5 mm in length.

To effective dose determine resulting from PET-CT tests to oncology diagnostic, was used two different methodologies, one to PET and other to CT.

Thus, the absorbed and effective doses from PET-CT were obtained with the algebraic sum of these quantities in PET and CT.

2.1. Absorverd and effective doses from ^{18}F-FDG PET

According to the model proposed by the ICRP 106, coefficients are used ($\Gamma_{T}^{^{18}F-FDG}$) can calculate the absorbed dose in organs (D_T) value from the injected radioactivity (A) and thus determine the patients effective dose (E) from the weighting factor for each tissue or organ (W_T), the weighting factor for the type of radiation (W_R) of radionuclide injected (F^{18}), and age of the patient, according to Equations 1 and 2:

$$D_T = \Gamma_{T}^{^{18}F-FDG} \cdot A \quad (1)$$

$$E = \Sigma\left[w_T \cdot \Sigma\left(D_T \cdot w_R\right)\right] \quad (2)$$

The activity of the radiopharmaceutical used for calculation was determined from the anthropomorphic phantom male weight. Was used the 3,7 MBq.kg^{-1} ou 0,10 mCi.kg^{-1}.

2.2. Absorbed and effective doses from computed tomography (CT)

The absorbed and effective doses patients received during the ^{18}F-FDG PET-CT examination exclusively from CT were evaluated by using thermoluminescent detectors inserted in anthropomorphic phantom Alderson Rando® interested (lens, thyroid, breast, brain, pituitary, brain, lung, heart, bladder, sigmoid colon, intestine, kidney, liver, gallbladder, pancreas, spleen, marrow, stomach and testicles).

Many points were selected based on organs mass and volume, were inserted three TLD-100 encapsulated detectors to increase the reliability of metrological measurements.

After their preparation, the anthropomorphic simulators were tested for patients similar images acquisition with cancer patients protocols and a single irradiation was performed.

As the CT image on ^{18}F-FDG PET-CT has the purpose of anatomic location, it follows the same protocol for all patients, clinical indication regardless.

Primarily a scout was conducted over a distance of 150 cm, using 120 kV electric current of 10 mA and rotation time of 1s tube, starting at the top of the head.

For CT acquisition, the starting point of data acquisition (skull base) is verified based on obtained image with the scout and was used 120 kV, automatic electrical current voltage ranging between 10 and 120 mA, thickness beam 40 mm section thickness of 0.625 mm, a pitch of 0.984 speed of 39.37 mm/rot, rotation time of

0.7s tube noise level of 25.00 with a 98.7 cm length scan in males phantom, ending at the root of the thigh, so only the region directly receives the primary X ray.

3. RESULTS

3.1. Absorbed and effective doses result from the ^{18}F-FDG PET

The absorbed and effective dose values from the merger of the radiopharmaceutical ^{18}F-FDG, were calculated considering the injected activity 272MBq.

The effective dose calculated to anthropomorphic phantom was 4.84 mSv.

The estimated absorbed doses are considered low when compared to dose thresholds for deterministic effects may cause in the organs under study.

It may be noted that the absorbed dose was increased in organs such as heart and brain, have high glucose metabolism, as expected. The bladder also shows higher absorbed dose values, since the 18F-FDG radiopharmaceutical excretion main form it is through the urinary tract.

Huang et al. (2009) used the same methodology to determine the effective dose from radioactivity of ^{18}F-FDG, but with fixed activity of 370 MBq for all patients undergoing a PET-CT and using the tissues or organs weight factors published by ICRP 80, obtaining effective dose of 6.2 mSv resulting from the incorporation of ^{18}F-FDG, 20.6% higher than the value obtained in this study.

Brix et al. (2005) used the same methodology of Huang et al. and estimated effective doses performed with two different protocols, but both with fixed injected activity in patients. The protocol was performed using 300 MBq an average per patient and protocol B 370 MBq. The estimated effective doses were 5.7 and 7.0 mSv, respectively, showing good agreement with the studies presented.

The ^{18}F-FDG activity incorporated calculation for each patient, variable with mass, favors the optimization of dose, conduct internationally recommended and required by national regulators, obtained by applying the basic radiation protection.

3.2. Results of absorbed and effective doses from computed tomography (CT)

The organs absorbed dose results evaluated in the Alderson Rando® male phantom using whole body oncology protocol with modulation of electrical current, noise index of 25 and 98.7cm scan.

The effective dose with modulation current was (3.67 ± 0.95) mSv. The expanded uncertainties of measurement were calculated taking into account the standard uncertainty of measurement multiplied by the coverage factor (k = 2), for a student-t distribution with effective degrees of freedom goes to infinity, corresponding to a coverage probability of 95.45% (ISO 2008).

The absorbed dose is highest in the testicle, thyroid and breast. All this organs are located directly in the irradiation field. However, when considering the organs radiosensitivity that most contribute to the effective dose breast, testicle and lung , respectively.

The uncertainty sources that more contributed to the expanded uncertainty in all effective dose calculations were thermoluminescent dosimeters reproducibility and calibration.

3.3. Effective dose from the PET-CT with low dose CT scan for diagnosis and stage cancer

Adding to the effective dose from incorporation of ^{18}F-FDG and the CT use for anatomic mapping, the average effective dose of PET-CT scans for diagnosis and staging cancer in male patients was (8.51 ± 2.21) mSv.

Huang et al. (2009) determined the effective dose using three different CT protocols in an equipment of 64 channels and ^{18}F-FDG activity injected in patients. The estimated effective doses were 13.65, 24.80, and 32.18 mSv for male patients and 13,45, 24,79 e 31,91 mSv for female patients.

The effective dose resulting from examination of ^{18}F-FDG PET-CT whole-body oncological diagnosis and staging evaluated was approximately 3 times lower than values in the other assessments. These differences are mainly the result of effective dose from CT. In the works cited, most CT protocols use diagnostic quality while at in this work used only for body structures anatomic location, without diagnostic quality.

4. CONCLUSION

Using the PET-CT image acquisition protocol for diagnosis and cancer stage, with the anatomical mapping CT protocol, the largest contribution to the effective dose is from the radiopharmaceutical, approximately 60% of the total effective dose.

ACKNOWLEDGEMENTS

The authors are thankful to Federal University of Minas Gerais, FAPEMIG and CNPq for supporting this work with fellowships and financial resources.

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3 years of clinical follow up of stress first myocardial perfusion scintigraphy (MPS) using IQ-SPECT system; reducing radiation exposure

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Introduction: Medical radiation from X-rays and nuclear medicine is the largest man-made source of radiation exposure in Western countries. It accounts for a mean effective dose of 3,0 miliSievert (mSv) per person per year, equivalent to the dose of 150 chest X-rays. To protect patients, doctors, technicians and nurses, exposure should not be more than necessary. Guidelines for Myocardial perfusion SPECT studies recommend that if stress examination is performed first and reported as normal, the rest examination can be omitted. Myocardial perfusion SPECT studies using IQ-SPECT Symbia S system (Siemens, USA) SMARTZOOM collimators and with dedicated reconstruction software provides shorter scan time using lower levels of radioactivity compared to the conventional SPECT systems. We evaluated prognostic value of stress only normal study in 3 years of clinical follow up compared to stress-rest normal study using IQ-SPECT.

Subjects and methods: We enrolled 606 patients (430 female age 45-75, mean 65, 176 male age 42-83, mean 66) with suspected coronary artery disease who underwent myocardial perfusion SPECT on February, March, April and May 2013 and evaluated as normal scans. Stress first (386 patients) and two days protocols (59 patients) were used. The patients' hospital data records were checked for follow up in three years. We excluded 161 patients who didn't visit our hospital in the following three years and who couldn't be reached by

phone. Myocardial perfusion GATED SPECT imaging results were compared with the clinical outcomes of the patients.

Results: Among the normal scans with stress first protocol 34 patients (8,8%) had a cardiac event/symptom or revascularisation. Among the normal scans with two days protocol 5 patients (8,4%) had a cardiac event/symptom or revascularisation. Statistical analysis results revealed no significant difference (p value<0,01) between the two studied groups' prognoses for 3 years.

Conclusion: Patients having normal stress imaging only have the same cardiac event rate in three years follow up as patients having normal stress-rest imaging. The stress-only protocol can be used with the IQ-SPECT gamma camera system reducing radiation exposure for patients as well as nuclear medicine department staff.

Accidental Exposure of Foetus during Imaging of Pregnant Patient at the Gynaeco Obstetric and Paediatric Hospital of Douala, Cameroon

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Abstract

The use of X-Ray in medical radiology is always risky. In particular, when it comes to imaging pregnant women, the majority of foetus cells can be damaged. Despite all the requirements from international renowned organizations relating to imaging of pregnant women and new-borns, the unintentional irradiations of the foetus are still common in radiological procedures. A case that occurred at the Gynaeco Obstetric and Paediatric Hospital of Douala (Cameroon) in September 2015, is instructive. A 46-year-old woman about six months pregnant was subject to an X-Ray of the pelvis requested by a medical doctor following a suspicion "of osteonecrosis ». Investigations conducted by the National Radiation Protection Agency (NRPA) revealed that the patient was irradiated with parameters of 20 mAs and 90kV which lead to a dose to foetus of about 213 µGy. The lead apron that was used to protect the pelvis reduced the absorbed dose by a factor of 6. The hospital officials were advised to do more collaboration between referral medical doctors and radiologists as soon as the use of X-Ray on pregnant women is deemed necessary, to inform patients about the dangers of ionizing radiation on foetus, and to request for information on their pregnancy status as well.

1 Introduction

One of the most frequently asked questions in relation to the use of ionizing radiation in medicine concerns the management of pregnant patient. Instinctively, one may want to avoid use of ionizing radiation with a pregnant patient. However, there are a number of situations in which the use of ionizing radiation for diagnosis or therapy is appropriate. In addition, there are many female physicians and technicians who are employed in medical practices using ionizing radiation. Thousands of pregnant women and radiation workers are exposed to ionizing radiation each year (ICRP 84, 2000). According to Annex D of UNSCEAR 2000, X rays have also been used for more than 50 years to assess the dimensions of the maternal pelvis in pregnancy. For occupationally exposed pregnant women, the equivalent dose to the surface of the abdomen shall not exceed 2 mSv per year and the effective dose resulting from exposure shall not exceed 1 mSv from the time which the pregnancy is known until its term (ICRP 60 1991; Arête N° 1152/A/MINSANTE of 2013). According to presidential decree N° 2002/250, issued on 31st October 2002 (Cameroon Official gazette 2003), National Radiation Protection Agency (NRPA) is the competent authority for radiation protection and waste management issues. In this regard, NRPA authorizes and inspects the use of ionizing radiation sources to protect people and the environment against the harmful effects of ionizing radiation.

Since the Decree N° 2002/250 states in its article 4 (5) that NRPA is responsible to respond to radiological accident/incident, that is why it was notified of the radiological incident related to the irradiation of the foetus with X-ray machine that occurred at the Gynaeco Obstetric and Paediatric Hospital of Douala, Cameroon. Following to this notification, NRPA team carried out an investigative mission to estimate the foetus dose of the pregnant patient.

2 Material and method

2.1 Management of a pregnant patient

Ministry of Public Health Order Number 1152/A/MINSANTE of 2013 prescribes that measures should be taken to manage pregnant patients in diagnostic radiology. All X-ray examinations shall be justified whether the patient is pregnant or not. In addition, posters and radiation trefoil should be posted at surveyed and controlled areas within diagnostic X-ray departments and areas where diagnostic X-ray equipment is used to avoid unintentional radiation exposures of the embryo and foetus. When a patient has been determined to be pregnant or possibly pregnant, the radiologist usually begins by determining whether the foetus is going to be in the primary X-ray beam. If not, then the risk to the foetus is extremely low and the most important thing is to keep the number and type of exposures to a minimum while still getting the correct diagnosis.

When an examination is indicated in which the X-ray beam irradiates the foetus directly, and this cannot be delayed until after pregnancy, the most common ways to tailor examinations and reduce foetal exposure are to collimate the beam to a very specific region of interest. When a high-dose procedure is performed and when the foetus is known to be in the primary X-ray beam, the technical factors should be recorded to allow subsequent foetal dose estimation.

2.2 Experimental Foetal dose estimation

NRPA investigation protocol has been used to measure entrance skin dose (ESD) of the pregnant woman. According to Mahadevappa in 2011, Foetal dose be conservatively estimated as 0.15 times the entrance skin dose (ESD) for conventional radiography and fluoroscopy techniques.

Figure1: DIAVOLT kVp-meter

DIAVOLT kVp-meter (figure 1) which is non-invasive kVp, Peak Potential Voltage (PPV), dose and time meter for acceptance tests and quality control (QC) of diagnostic X-ray equipment was used for this measurement. According to the NRPA Guidance N°0050 (2016) on quality control of X-ray machine in diagnostic radiology, the given steps below were followed:

- Mode RAD/FLU was chosen;
- kVp ranged from 40 to 150 kV was set up ;
- filtration of 2.5 mmAl as indicated on the tube was selected;
- DIAVOLT kVp meter was positioned on the top of the table at 100 cm of X-ray tube;
- Laser light field was collimated within a standard size of the DIAVOLT;
- 20mAs and 90 kV were chosen for the tests;
- three measurements of ESD were recorded.

The DIAVOLT kVp meter was covered by a lead apron of 0.25 mmPb to simulate a similar condition of the examination for which the following physical parameters were used:

- Source to image receptor distance of 1meter;
- Tube potential setting is 90 kV;
- Tube current setting 200 mA;
- Exposure time 0.1 second;
- Beam size: 43 cm x 35.5 cm;
- Patient AP thickness is 26 cm;
- Total filtration 2.5 mmAl.

Table 1: Machine Characteristics

	Generator						
	Supplier	**SN**	**Model**	**Filtration**	**homologation Number**	**Year of fabrication**	
WDM Radiology Equipment	China RessourcesWandong Medical Equipment Co.Ltd	10320-91	GFS50 2-8	/	/	2012	
	Tube						
	Supplier	**SN**	**Model**	**mA Max**	**kV-Max**	**Filtration (mmAl)**	**homologation Number**	**Year of fabrication**
	China RessourcesWandong Medical EquipmentCo.Ltd	1205141	DX52-30.50/125H-T2F	/	125	2,5mmAL	/	2012

3 Results and discussion

Table 2 shows measured doses by Diavolt kVp meter covered with lead apron. The mean value of about 1420 μGy was obtained.

Effectiveness of the shielding enclosed in lead apron was appreciated through a ratio between measured doses by Diavolt without a covered lead apron. Therefore, the shielding used to cover the pelvic attenuated the direct beam by a factor of about 6. The mean dose of 1420 μGy measured by Diavolt with a lead apron on it was used to estimate the foetus absorbed dose according to Mahadevappa in 2011. The obtained value which is 213 μGy is less than 100 mGy above which malformations may be suspected. This dose is relatively low and cannot be responsible for malformation effects. However, it presents a minor risk of cancer and leukemias for children aged from 0 to 15 years whose mothers have undergone irradiation during pregnancy.

Table 2: ***Dose Measurements by Diavolt covered with lead apron***

Parameters	**Essay N°**	**Measured Entrance Skin dose (**μGy)	**Fetus dose estimated (ESD*0.15) (**μGy)
	1	1420	-

20mAs,	2	1421	-
90kV	3	1419	-
	Mean	**1420**	**213**

According to the responsibilities of practices, the risk of cancer in teenagers must be taken into account in the follow-up of the pregnant woman by emergency physician and the radiologist. The need to justify any X-ray examination in this case and possibly to use non-irradiating imaging with equal diagnostic performance is advised. If the proposed radiological examination is the only way to establish the necessary diagnosis for the appropriate examination management, emergency physician and the radiologist are required to inform the patient of the risks of malformations and cancers that can occur.

The radiologist and radiation protection officer are responsible for recording the estimated dose of irradiation on the examination report. The medico-legal responsibility of the interveners imposes in addition to the justification, taking into account prenatal and postnatal risks.

4 Conclusions

In general, the medical follow-up of pregnant women is delicate and requires the collaboration of all the stakeholders. The ignorance of the effects of ionizing radiation on the foetus by some medical staff and the insufficient collaboration between them can be considered as the main causes of the incident. At the same time, the irradiation of a pregnant woman must be the subject of a documented consensus between medical staff. A report specifying the dose received in the abdomen must be recorded by the radiologist and radiation protection officer. The patient's information on the risks that can occur during the irradiation is prior to the X-ray exam is archived.

According to this incident, it was recommended to diagnostic radiology in medical sector in Cameroon to:

- justify any X-ray examination required for a pregnant woman when non-irradiating imaging cannot be used ;
- establish mechanisms for collaboration between medical personnel involved in the follow-up of pregnant women in order to take into account effects of ionizing radiation on the foetus and / or teenage year;
- put in place mechanisms for informing pregnant women about the risks to the foetus and / or teenagers when it is undergoing irradiation;
- put in place mechanisms for early notification of incidents;
- carry out training on radiation protection of personnel involved in radiology.

These measures are now shared during implementation of NRPA inspection program in the whole medical sector in Cameroon.

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IN HOUSE FULLY AUTOMATED PRODUCTION OF F-18 FLORBETAPIR PET TRACERS ON THE CFN-MPS200 RADIOSYNTHESIZER WITH CUPID SOFTWARE MODIFICATION

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Abstract

Nowadays, F-18 florbetapir is a PET radiopharmaceuticals for diagnostic Alzheimer's disease. In Siriraj Cyclotron Centre has produced F-18 florbetapir using semi-auto radiosynthesis. To minimize the staff exposure to ionizing radiation from manual purification procedures, a fully automated radiosynthesis production method was developed. Results from the fully automated synthesis yield of F-18 florbetapir was 25±2.7% with a RCP>95%. Radiation exposure in radiopharmaceuticals room (monitored with MediSmarts radiation monitoring system) and the accumulated dose of staffs (collected from personal radiation monitor) were significantly decreased. In conclusion, the fully automated radiosynthesis can help to reduce occupational radiation exposures for staff and suitable for use in clinical PET imaging.

1. INTRODUCTION

F-18 Florbetapir is a F-18 labelled PET tracer that specifically binds to amyloid-β (Aβ) peptide in amyloid plaque. Detection of amyloid plaques is one of diagnostic keys of Alzheimer's disease (AD) and accumulation of the Aβ plaques in the brain is the most significant factors associated with the development of AD [1-5].

Regarding the production of F-18 Florbetapir, an instant fully-automated radiosynthesizers has been developed under the license of Eli Lilly and Company [6-8]. Some institutes may generate their own software to support their radiosynthesis procedures, so called in house software. Major advantage of this kind of software is its flexibility. For instance, it allows user to create specific program or reconfigure the tubing layout for generating new radiotracer.

For the purification process of F-18 Florbetapir, a semi-preparative HPLC (High-performance liquid chromatography) equipment is generally used [9]. This process is quite complicated and takes approximately 60 minutes for entire process. This drawback leads to development of an automated synthesis, which provides a rapid processing and automatic purification system that help decreasing radiation exposure during the routine procedure.

With several advantages of fully-automated radiosynthesizers mentioned above, we try to create our in-house software to decrease complication of working process and radiation exposure to our staffs.

2. EXPERIMENTAL

We used CUPID software to create a new schematic diagram and a program for two CFN-MPS200 synthesizers. Our in-house fully automated radiosynthesis system of F-18 Florbetapir composed of solenoid valves for control the direction, vacuum pump, regulated pressure sources, evaporator for remove the solvent, HPLC semi-preparative system for purification, UV detector, radiation detector and temperature controllers.

Process of automated radiosynthesis began with absorption of F-18 ions in an anion exchange column on the equipment and introduction of F-18 ions into the reactor. Then, reactor was heated to remove the solvent from the solution and acetonitrile (ACN) was introduced into the reactor. After that, the reactor was reheated to sufficiently dry the inside of the reactor (azeotrope). Following azeotropic distillation, the acetonitrile/dimethyl sulfoxide solution containing the raw material (AV-105) was introduced into the sufficiently dried reactor. AV-105 was heated to label with F-18 (fluorination step) afterward. After fluorination, 1 N hydrochloric acid (HCl) was introduced into the reactor to hydrolyze the protecting groups of the raw material. The reaction mixture was then adjusted the pH with 1 N sodium hydroxide (NaOH) and finally, the raw material became F-18 Florbetapir ***Fig. 1-2.***

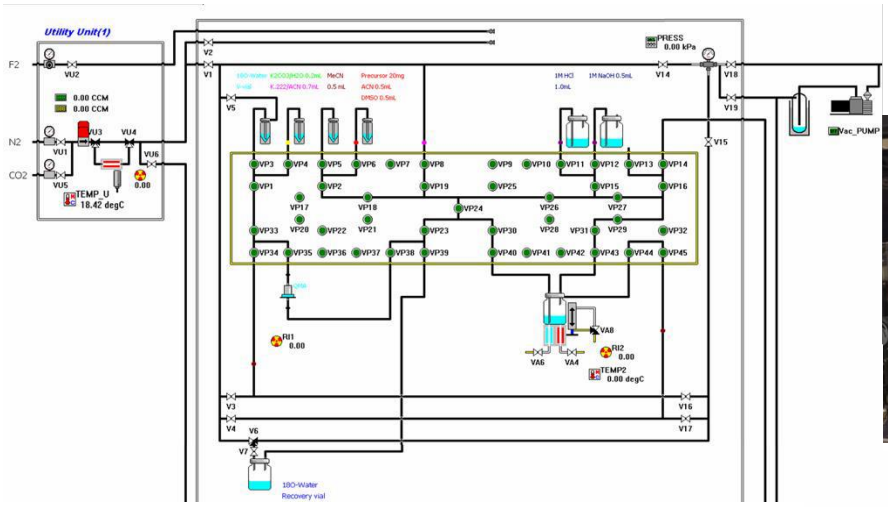

Fig. 1 *Diagram of the automated radiosynthesis of F-18 Florbetapir.*

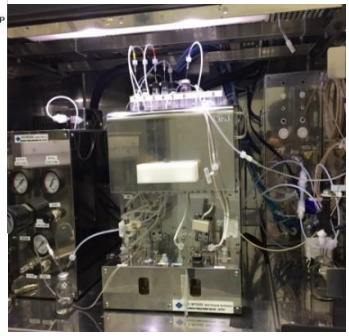

Fig. 2*. CFN-MPS200 module for F-18 Florbetapir.*

Following the radiosynthesis procedure, the obtained liquid containing various impurities was subjected to purification process. After a neutralizing solution was introduced into the reactor to neutralize reaction liquid, the liquid was transferred to the HPLC injection unit, where the isolation process occured. F-18 Florbetapir, the product substance, was collected in a pear-shaped flask, whereas impurities were abandoned. Next, F-18 Florbetapir was diluted with sterile water before passed through C18 Sep-Pak cartridge. The F-18 Florbetapir was then eluted by using ethanol (EtOH), heated to evaporate EtOH and neutralized with 0.9% sodium chloride injection. Finally, a unit dose of F-18 Florbetapir was dispensed into a vented sterile collection vial by passing through a 0.22 µm sterile filter (PVDF) ***Fig. 3-4***.

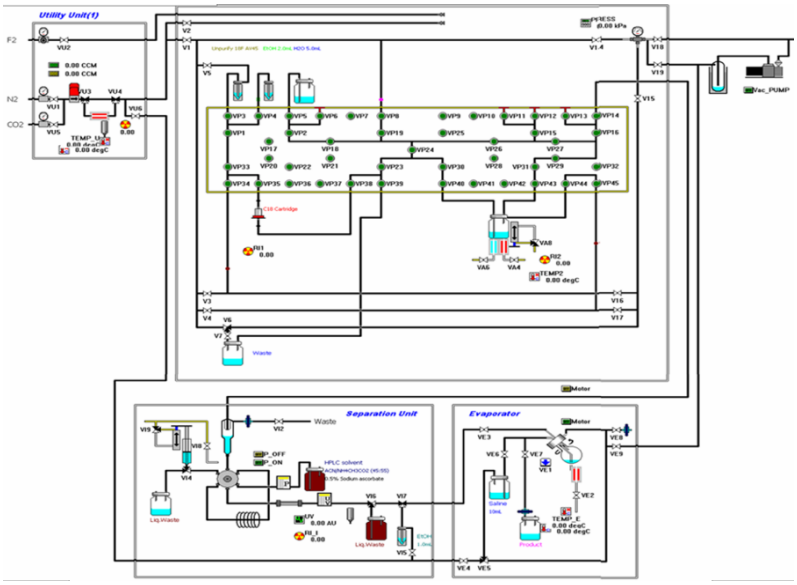

Fig. 3. *Diagram of the automated purification of F-18 Florbetapir.*

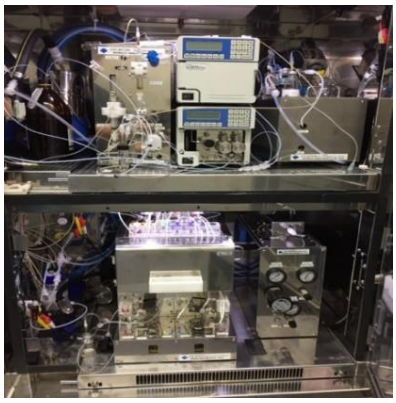

Fig.4. *CFN-MPS200 module and automated purification system.*

Yield and RCP were measured based on 12 productions whereas radiation exposure were collected based on 17 productions. Radiation exposure in workplace was monitored by MediSmarts radiation monitoring system and Geiger-Muller counter (GM counter). There were total 11 monitored spots including hall way, control room, cyclotron room, electric room, quality control room, preparation room, hot cell room and 3 exhaust system as shown in ***Fig. 5.*** Level of radiation exposure was recorded in range and each range was demonstrated in colored lights to make it easy to be recognized. All 5 colored light representing range of exposure were white, blue, green, orange and red for radiation in range of 0-1 µSv/h, 1-19 µSv/h, 20-24 µSv/h, 25-99 µSv/h and more over 100 µSv/h, respectively

The working area of Siriraj Cyclotron Centre was divided into 3 areas depending on the radiation level:

(I) High radiation areas were places where radiation exposure could be higher than 50 millisievert per year (mSv/y) or 25 microsievert per hour (uSv/h). This area included cyclotron room and hot cell room.

(II) Medium radiation areas were places where radiation exposure could be higher than 15 mSv/y or 7.5 uSv/h. This area included quality control room and preparation room.

(III) Low radiation areas were the other rooms in the vicinity of cyclotrons, including hall way, control room and electric room.

Occupational exposure was monitored in 6 staffs who are responsible for production and distribution, quality control and medical physics.

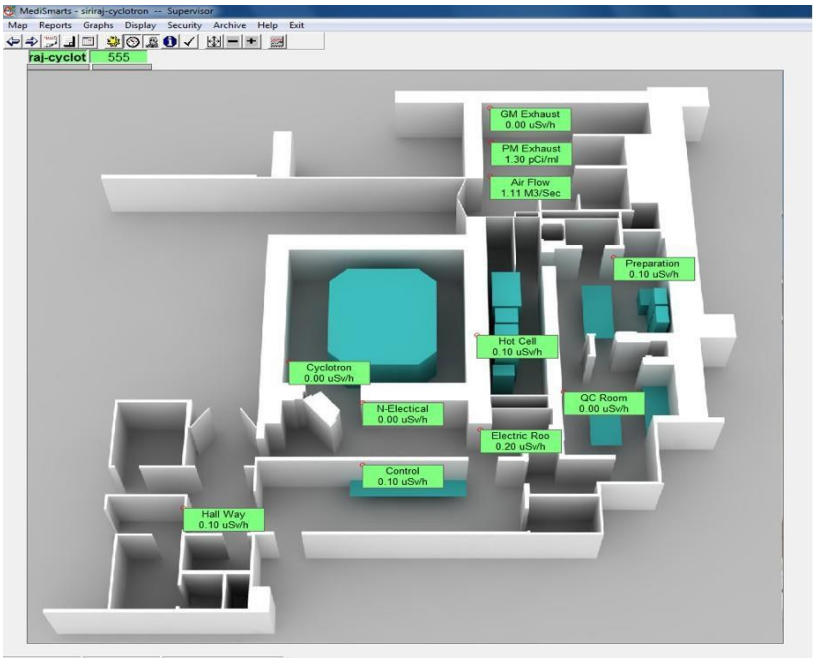

Fig. 5. *Monitored spots for radiation exposure in Siriraj Cyclotron Centre.*

3. RESULT

We successfully developed an in house fully automated syntheses, composing of F-18 Florbetapir synthesis, HPLC purification and solid phase extraction procedures. F-18 Florbetapir was obtained with RCP of over 95%, radiochemical yield of 25.0±2.7% in the quantity of 400-500 mCi (14800-18500 MBq) and total procedure time of 90 minutes decreasing from that of 120 minutes with manual purification.

For manual purification, radiation exposure for low, medium and high radiation area of the Siriraj Cyclotron Centre were 0.1-0.2 µSv/h, 1.0-2.0 µSv/h and 25.0-100.0 µSv/h, respectively. When using automatic purification, radiation exposure of low, medium and high radiation area were 0.1-0.2 µSv/h, 1.0-2.0 µSv/h and 1.0-5.0 µSv/h, respectively. Radiation exposure in low and medium exposure areas were not different between manual and automatic purification method, but is significantly different for high exposure area as shown in Table 1.

TABLE 1. THE RANGE OF GAMMA RAY DURING F-18 FLORBETAPIR BETWEEN USING MANUAL PURIFICATION AND AUTOMATIC PURIFICATION

Radiation Level	Manual purification* (μSv/h)	Automatic purification* (μSv/h)
Control room+	0.1-1.0	0.1-1.0
QC room and Preparation room++	1.0-2.0	1.0-2.0
Hot cell room+++	25.0-100.0	1.0-2.0

*The average of radiation exposure for each radiation area; +low, ++medium and +++high level radiation are.

With semi-automated synthesis, radiation exposure for production and distribution staff, quality control and medical physics were 68.80 μSv/h, 2.20 μSv/h and 1.58 μSv/h, respectively. The highest exposure was observed in production and distribution staff. When using automated synthesis, radiation exposure of production and distribution staff, quality control and medical physics staff were decreased 92.09%, 6.82% and 1.90%, respectively as shown in Table 2.

TABLE 2. RADIATION DOSE OF STAFF MEMBERS FROM F-18 FLORBETAPIR PRODUCTION PER TIME

Jobs	Using semi-automated syn. (μSv/h)s	Using automated syn. (μSv/h)
Production & Distribution	68.80	5.44
Quality control	2.20	2.05
Medical physic	1.58	1.55

4. DISCUSSION

We achieve F-18 Florbetapir production with our own in house fully automated synthesis software. The labelling yield of 25.0±2.7% by an in-house software is lower than that of 40-45% by commercial standard. However, we have only 3-4 patients per day and obtained radiotracer per production is adequate.

The remarkable benefit of this in-house software is significant reduction of radiation exposure. This software reduces radiation exposure by two means; 1) time of purification process is shortened from 60 minutes to 30 minutes, and 2) the entire process of production becomes a close system when using a fully automated synthesis software. As a closed system, there is no more manual transfer of radiotracer from radiosynthesis system to purification system. This leads to marked reduction of radiation exposure to our staff (92.09%). In addition, contamination in hot cell room is decreased to a very low level, making production of other radiopharmaceuticals immediately after finishing production of F-18 Florbetapir possible.

At this moment, this in-house software works well with our services. Even so, we have plan to develop the software to support an external controller in the future. With an external controller, we should be able to monitor synthetic processes from control room resulting in further reduction of radiation exposure.

5. CONCLUSION

An in-house software created to support the production of F-18 Florbetapir at the Siriraj Cyclotron Centre is currently in operation for routine clinical production of this important PET radiotracer. Adequate amounts of F-18 Florbetapir can be routinely produced for clinical service, in a significantly reduced total time of production and significantly reduced radiation exposure to personnel involved in the radiotracer production and dose formulations process.

Further information

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ESTIMATE ANNUAL RADIATION DOSE TO STAFF PER NUCLEAR MEDICINE EXAMINATION

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Abstract

Nuclear medicine is a medical that involves the study distribution and localization of the administered radiopharmaceutical to provide functional or metabolic information. Therefore, patients and staff are subject to radiation exposed during nuclear medicine imaging procedure. The potential radiation dose offered from the nuclear medical examinations, the amount of activity and the different condition to receiving radiation doses while performing procedures such as preparing the patient on the gamma camera bed, monitoring the patient during data acquisition, escorting the patient to the department and waste management form the patient. Knowing the radiation doses from patients to staff is very important because it leads to workflow and radiation safety. The aim of this study to estimate annual radiation dose to staff per nuclear medicine examination at Nuclear Medicine department, Maharat Nakhon Ratchasima hospital, Thailand. In this study, we measure the radiation dose by pocket electronic dosimeter from 106 patients undergoing nuclear medicine examination. 50 patients were measured with bone scan, 4 underwent myocardial perfusion, 14 underwent venoscintigraphy, 20 underwent thyroid function, 4 underwent renal function study, 2 underwent lower GI bleeding study, 7 underwent lung perfusion and 5 underwent total body scan post I-131 therapy. At the surface body from the patient, the radiation doses ranged from 79.58 ± 20.4, 42.69 ±3.94, 34.82 ± 7.11, 27.95 ±10.00, 4.87 ± 0.95 , 4.1 ± 0.85 and <1 µSv per hour respectively. By comparing the amount of radiation that a worker would receive annually per examination are as follow 64.85, 2.94, 2.95, 27.45, 1.03, 0.06 and < 1 mSv. The total radiation dose was 99.31 mSv per year. Conclusions and Recommendations: The study radiation dose that a worker would have received from patients can be used as a basis to estimate the radiation dose from the patient to the environment. In addition, and more important these results could be used to modify the process to work and improve staff education. Moreover, it should be have the separation area for patients who receive radionuclide.

1. INTRODUCTION

Nuclear medicine is the branch of medicine that involves the administration of radioactive substances in order to diagnose and treat disease. Patients and nuclear medicine staffs are subject to radiation expose from the radiopharmaceuticals administered to a patient for diagnostic or treatment purposes [1,2]. The potential radiation dose offered from the nuclear medical examinations depend on the amount of radioactivity and the different condition to receiving radiation doses while performing procedures such as preparing and administering the radiopharmaceutical, positioning the patient on the gamma camera bed, monitoring the patient during data acquisition, escorting the patient to the department and waste management from the patient [3].

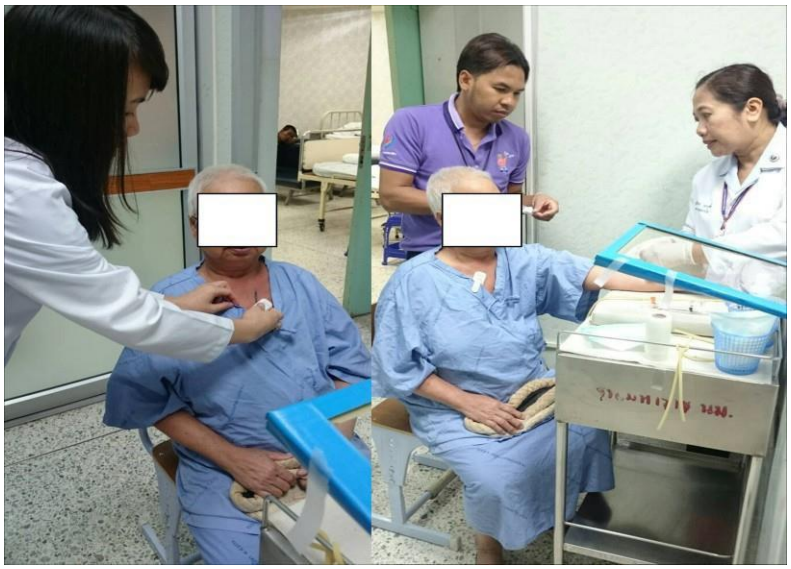
To knowing the radiation doses from the patients to staff is very important because it should be help in determining the rotation time of staffs in different procedures and differences in their individual technique for radiation safety. Thus, at Nuclear Medicine department, Maharat Nakhon Ratchasima hospital,Thailand, all staffs routinely use pocket electronic dosimeter to monitor the radiation dose per day. The aim of this study to estimate annual radiation dose to staff per nuclear medicine examination at Nuclear Medicine department, Maharat Nakhon Ratchasima hospital and to determine dose rates from patients for the most common diagnostic procedures.

2. MATERIAL AND METHODS

Normally, direct experimental determination of the external radiation dose to the nuclear medicine staffs per single procedure may follow 1 of 2 strategies. The first is based on accurate measurements of the dose rate at fixed distances from the patient and less accurate evaluations of the time spent by the operator at those distances. The second strategy consists of the direct reading of pocket electronic dosimeter by the staffs during the procedure. In this study used direct measurements from the patient, although it is a rough approximation of dose rate but is more general and allows a direct comparison of dose rates between different published data [4,5]. Thus, we measure the radiation dose by pocket electronic dosimeter at the body surface of 106 patients undergoing nuclear medicine examination as show in **Fig 1**. 50 patients were measured with patient underwent bone scan, 4 underwent myocardial perfusion, 14 underwent venoscintigraphy, 20 underwent thyroid function study, 4 underwent renal function study, 2 underwent lower GI bleeding study, 7 underwent lung perfusion and 5 underwent total body scan post I-131 treatment. Then, radiation doses to staffs worker in nuclear medicine were estimated by calculating the time interval and calculated dose rate from the patient at various distances 0.5 and 2.0 m. In this estimation, the distance of the staff from the patient was closed to the patient (body surface) during positioning on the gamma camera bed and waste management of patient. In some cases, patients positioned themselves after receiving instructions. The staffs stood 0.5 m from the scanner bed while they were

speaking with the patients about the procedure. Operators carefully measured the mean time spent by staff at these distances and using a personal pocket electronic dosimeter.

FIGURE 1. POSITIONING OF POCKET ELECTRONIC DOSIMETER ON PATIENT BODY SURFACE BEFORE INJECTION RADIOPHARMACEUTICALS



For thyroid scans, the patients were imaged using a gamma camera after the intravenous injection of 2-5 mCi of ^{99m}Tc-pertechnetate. Imaging began 15-20 min after the injection.

For perfusion lung scans, the patients were imaged immediately after the intravenous injection of 5 mCi of ^{99m}Tc-MAA. The standard 8 views static of lung; anterior posterior, both oblique, both lateral were obtained on a dual-head planar gamma camera and in some case performed SPECT-CT.

For whole-body bone scans, the patients were imaged 2-4h after the intravenous injection of 15-20 mCi of ^{99m}Tc-MDP. The standard anterior and posterior whole-body images, anterior and posterior pelvis and both lateral skull were obtained on a dual-head planar gamma camera.

For ^{99m}Tc-MAG3, ^{99m}Tc-DTPA renal function scanning, the activity was 3-5 mCi. In most cases, the patient will be positioning on the gamma camera bed first, and then imaging dynamic study together with intravenous injection.

For ^{99m}Tc-MAA, ^{99m}Tc-Phytate venoscintigraphy, the activity was 10-15 mCi. Patient will be lie on the gamma camera bed, and then the radiopharmaceuticals were bipedal injection. The standard anterior and posterior whole-body imagings of on and off tourniquet were obtained.

For ^{99m}Tc-RBC lower GI bleeding study, the stannous was injected intravenous 0.03 mg/kg, and then 20 minutes later the ^{99m}Tc-pertechnetate 15-20 mCi will be injected. Imaging began 5 minutes after, the standard anterior and posterior abdomen images were obtained every 5 minutes until 1 hour and then 2, 4, 24 hours respectively.

The procedure for myocardial perfusion scanning with ^{99m}Tc-MIBI had 2 stages. The first involved stress imaging after injection of 5-7 mCi of ^{99m}Tc-MIBI at peak exercise. Gated SPECT-CT was performed from 1 h after the injection. The second stage involved rest imaging after administration of 15-20 mCi of ^{99m}Tc-MIBI. Gated SPECT-CT was performed within 1 h after the injection. Likewise, the procedure for myocardial perfusion scanning with ^{201}Tl-chloride had 2 stages, the first involved stress imaging after injection of 3-5 mCi of ^{201}Tl at peak exercise. Imaging began within 10 minutes after the injection. The second stage will be imaging 3-4 hours later for redistribution stage.

3. RESULTS

The radiation doses ranges at body surface of the patient were 79.58 ± 20.4, 42.69 ±3.94, 34.82 ± 7.11, 27.95 ±10.00, 4.87 ± 0.95, 4.1 ± 0.85 and <1 µSv per hour respectively. By comparing the amount of radiation that a worker would receive annually per examination are as follow 64.85, 2.94, 2.95, 27.45, 1.03, 0.06 and < 1 mSv. The total radiation dose from the patient was 99.31 mSv per year as show in table 1.

TABLE 1. EXTERNAL DOSE RATES PER EXAMINATION MEASURED DIRECTLY FROM THE PATEINT AND THE ESTIMATION OF ANNUNAL RADIATION DOSE TO STAFF PER NUCLEAR MEDICINE EXAMINATION AT VARIOUSDISTANCES

Type of examination	No. of patient	Radiation dose rate per examination at various distance (μSv)			Estimate annual radiation dose	
		Body surface	0.5 m	2 m	No. of patient / year	Radiation dose (mSv)
Whole-body scan	50	79.58±20.4	3.18	0.00199	815	64.85
Myocardial perfusion scan	4	42.69±3.94	1.70	0.0010	69	2.94
Venoscintigraphy	14	34.82±7.11	1.39	0.00087	85	2.95
Thyroid function study	20	27.95±10.0	1.11	0.00070	213	27.45
Renal function study	4	4.87±0.95	0.19	0.00012	982	1.03
Total body scan post ^{131}I Rx	5	<1	-	-	235	-
Lung perfusion scan	7	<1	-	-	77	-
Lower GI bleeding study	2	4.1±0.85	0.16	0.00010	15	0.06
Infection study	1	1.58	0.063	0.00004	12	0.018

4. DISCUSSION ANDCONCLUSIONS

This study found that the external radiation doses to staff by reading the personal pocket electronic dosimeter were within permissible levels, regardless of whether a staff performed only a particular diagnostic procedure. Although, by comparing the amount of radiation that a staff would receive annually per examination were excess permissible levels if they are exposing directly from the patient but in practically, the staff no directly exposing radiation from the patient. However, the study radiation dose that a worker would have received from patients can be used as a basis to estimate the radiation dose from the patient to the environment, duration time for staff can be perform together with the patient. In addition, more important these results could be used to modify the process to work and improve staff education. Moreover, from the results could be applied to the restricted area for patients who receive radionuclide.

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RADIATION DOSE FROM NUCLEAR MEDICINE PATIENTS TO A NUCLEAR MEDICINE TECHNOLOGIST: RELATION TO ICRP RECOMMENDATION FOR PREGNANT WORKERS

Radiation dose from patients to technologist

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Abstract

The ICRP has recommended that the dose limit to the fetus during the declared term of pregnancy should not exceed 1 mSv. In order to verify, a direct measurements of the radiation doses to technologists carrying out a variety of imaging studies were made. From the studies, it is observed that in case of Tc-99m as a radioisotope, the dose to the technologist were below the detectable limit but in case of I-131 as a radioisotope, the radiation dose varied from 1-2 μSv per study. The study shows that there is unlikely to be a need to change the duties of technologists to remain within the proposed new dose limits. However, these findings should be helpful in allaying any anxiety.

Key words: radiation dose, imaging procedure, dose limit, ICRP recommendation

1. INTRODUCTION:

The International Commission on Radiological Protection (ICRP) has recommended that the dose limit to the fetus of the pregnant lady during the declared term of pregnancy should not exceed 1 mSv. In order to verify, direct measurements of the radiation dose to technologist carrying out a variety of imaging studies are discussed in this paper. Data has previously been published on the radiation doses received by nuclear medicine technologist in rendering necessary help to patients during imaging procedure (1,2,3). These were based on dose rates from patient's and time spent by technologist at different distances from the patient (4). Altering their work solely because of radiation exposure from patients would be exceptional (5).

The ICRP has recommended that the fetus should be treated as member of the general public (6). The relevant dose figure therefore becomes 1 mSv in a year, with the proposed supplementary dose limit being 2 mSv to the surface of the mother's abdomen during the declared term of the pregnancy. The decision is not due to any newly discovered risk to the fetus, but merely due to its reclassification as member of the general public.

Direct measurements were made on technologist to assess the dose received from patients undergoing a variety of nuclear medicine studies. These have been interpreted in the light of the new ICRP recommendations.

2. MATERIAL AND METHODS:

Digital personal pocket dosimeters (Bleeper Sv, Gothicrellon Ltd. Wokinghaam-England make) were used, which were calibrated to measure a minimum dose of 1 μSv. These were calibrated in free air geometry by using Co-60 source whose output in terms of exposure rate has been standardised by a reference standard graphite ion chamber. During patient investigations the readout was recorded both at

the start and at the end of the study. In addition to this, the patient's general condition, the experience of the technologist doing the study and any additional views, required by the reporting doctor was noted.

3. RESULTS:

Two pocket dosimeter were used in these studies, these dosimeter were given to the technologist before the start of the study and were collected back after the completion of the study. These studies were carried out in one of the major institutions in the country where, on an average 20-30 GBq/wk of Mo-99 radioisotope is used, in addition to other commonly used radioisotopes. In each study more than 15 patients were taken.

Table 1. shows the type of study carried out, age of the patient, the number of patients studied, the administered activity, the corresponding average radiation dose to the technologist as recorded by the dosimeter while performing the study, the time spent by technologist with patient per study and the equipment used for carrying out the study.

From the studies, it is observed that in case of Tc-99m, the doses to the technologists were below the detectable limit but, in case of I-131 as a radioisotope, the dose varies from 1-2 μSv per study. All the patients were fit mobile out patients. In some of the studies Dual Head Gamma Camera was used where after positioning the patient, technologist can perform the acquisition of images in the other room where computer console and other accessories were kept, introducing a distance factor in reducing the dose to technologist.

4. DISCUSSION:

An initial calibration check was performed to ensure that there was no systematic error in reading by any of the dosimeters. In our study, we have assumed that the duration of the pregnancy from the time the technologist informs her employer is 7 months, which is a conservative figure since a women do not usually work up to the time of delivery. In case, the women works up to the time of delivery i.e. 9 months then it is recommended that total radiation burden to be measured and work be stopped, if, total radiation burden approaches the recommended limit. In these studies nuclear medicine technologist did not receive any recordable radiation dose. Using nationally available data, cumulative doses can be calculated as a product of the number of studies per year and the radiation dose received per study.

Although it is proposed by ICRP (6) that all the dose limits to be reduced, the disproportionate reduction for pregnant women is not due to a newly described risk. It has been decided that a fetus should be treated as member of general public and that the mother cannot consent on behalf of her child to radiation above this level. There are, of course, assumed risks with any radiation exposure (7). The fatal cancer risk associated with 1 mSv exposure of the fetus is estimated to be 1 in 50,000, with an additional genetic risk of 1 in 100,000. Such risks are very small. Also, there is an evidence from the survivors of Hiroshima & Nagasaki of mental retardation when the fetus is exposed 8^{th} and 15^{th} weeks of pregnancy. The risk factor is 1 in 2500 per mSv, but this effect seems only to occur above 100 mSv. The above risk factors are considered to be small in comparison to the surprisingly high natural risk of 1 in 25 of any abnormality.

If we assume that the nuclear medicine technologist carries out 10 studies per day, 5 days a week for 7 months and also, if we consider 1μSv the maximum dose received per study then the cumulative dose received by the technologist is less than the recommended limit. In our study, we have only considered the dose to the nuclear medicine technologist from patients undergoing imaging studies. In addition, in some of the nuclear medicine centres, technologists undertake radiopharmacy work and/or injections also. The additional dose burden of these activities must be taken into consideration if this is the case.

5. CONCLUSION:

The study shows that there is unlikely to be a need to change the duties of technologists to remain within the proposed new dose limits. However, with reference to ALARA principle of ICRP, the pregnant lady may be given non-active work, if the authorities can find out suitable work for them.

TABLE 1. RADIATION DOSE TO TECHNOLOGIST IN μSV FOR EACH TYPE OF STUDY

Sr. No	Type of study	No. of patients	Age of patient	Administered activity	Dose to technologist	Time spent with patient per	Type of equipment
1.	Tc-99m–liver Scan	10	28-62 years	111-185 MBq	Below Detectable Limit	8-10min.	Single Head Gamma Camera
2..	Tc-99m Gastroesophagal reflux study	17	19 days – 9 months	18.5 MBq	-do-	20-25 min.	Single Head Gamma Camera
3.	Tc-99m-DMSA Renal Scan	12	2 months-15 years	111-185 MBq	-do-	10-15 min.	Single Head Gamma Camera
4.	Tc-99m Hepatobiliary Scan	12	2 months – 3 years	185 MBq	-do-	10-15 min.	Single Head Gamma Camera
5.	Tc-99m-Renogram with DTPA	11	3-61 years	74-185 MBq	-do-	30-40 min.	Dual Head Gamma Camera
6.	Tc-99m–Whole body	11	22-60 years	555-740 MBq	-do-	20-25 min.	Dual Head Gamma Camera
7.	I-131-Whole Body Scan*	10	28-60 years	2.7-6.66 GBq	1-2 μSv	30-45 min.	Whole Body Rectilinear Scanner

Note: * This study is being carried out after discharge of patient from the isolation ward.

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IMAGE QUANTIFICATION FOR DOSIMETRY EVALUATION IN THERAPY: URUGUAY INTERCOMPARISON RESULTS

Results of the participation of Uruguay in the intercomparison organized in the Coordinated Research Project E2.10.07, "Development of Quantitative Nuclear Medicine Imaging for Patient Specific Dosimetry".

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Abstract

In the frame of the Coordinated Research Project E2.10.07, "Development of Quantitative Nuclear Medicine Imaging for Patient Specific Dosimetry" an intercomparison work was done between 9 participating countries. The results of the second trial of the Uruguayan group are presented in this communication.

Images were acquired in a Nucline Spirit DH-V, MEDISO gamma camera with high-energy collimator. Planar and SPECT images were acquired from the sources inserted into a Jaszczak phantom filled with water. Four calibrated 133Barium sources provided by NIST were used as surrogate of 131Iodine. The images were acquired in multiple energy windows to apply the Triple Energy Window (TWE) method for scatter correction. One of the sources, was measured to calculate the calibration factor, in cps/MBq in gamma camera using the calibration value, as measured by NIST. These values were compared with measurements acquired from the quantification of both Planar and SPECT images. Activities were calculated from the observed count rates and the sensitivity factor and compared with reference activities. The ratio between the mean measured values that were reported in the intercomparison and the reference calibrated values reported by NIST was in the mean of the other countries being 0.91for planar images and 1.18 for SPECT.

1. INTRODUCTION

The use of targeted radionuclide therapies in cancer treatment is an important alternative to conventional therapeutical regimens. Radiopharmaceuticals are specifically developed to selectively deliver cytotoxic radiation emitting radionuclides to tumour cells, resulting in cell death. Comparing chemotherapy or beam radiation, targeted radionuclide therapy can avoid producing collateral effects in healthy tissues or organs. Alpha or beta particles and Auger electron emitting radionuclides may be used to label certain molecules such as antibodies or receptor avid peptides for targeted radionuclide therapy. High energy particle emitting radionuclides such as 131I, 188Re and 90 Y appear as appropriate for the treatment of large tumour burdens whilst, low and medium energy radionuclides like 177Lu and 153Sm seem to be more suitable for the treatment of small tumours or metastasis. The fundamental purpose of internal dosimetry is to predict response and toxicity of cancer therapy with internal emitters. A long experience in cancer treatment showed that absorbed dose can be used as a useful tool to predict biological response. [1-2]

Internal dosimetry calculations are based in the quantitation of the uptake of injected activity in various organs over time. The accuracy of these measurements is particularly important to rigorously optimize the therapeutic use of radiopharmaceuticals. As a consequence, there is a need for harmonized protocols or guidelines for acquiring quantitative information from nuclear medicine imaging procedures. 131I has 8 days of half life, presents gamma and beta emissions, the latter are the ones which produce the therapeutic effect because of its

high LET and low penetration power destroying the iodine up taking cells. When 131I is administered it goes straight to the thyroid and deposits all the energy causing cell death. Although administering high doses in the first stage gives better results, secondary effects may appear especially in the red marrow. Due to its high radiosensitivity any structural or functional alterations will affect blood formation with lesions such as aplastic anaemia. Other organs that may be affected are lungs, particularly if they carry metastasis, fibrosis can be developed. More than 30% of the patients treated with 131I may develop sialoadenitis due to irradiation. The risk of undesired effects can be controlled by a proper dose estimation of the administered dose, taking red marrow as the critical organ which cannot receive more than 2 Gy to avoid blood cells depression. Taking into account that 131I is the most widely used in therapies primarily associated with malignant thyroid diseases and despite their prolonged use, patients in developing countries are mostly treated with fixed doses without prior planning, however individual dose estimation ensures less side effects and more effective therapy. One of the possible methodologies to determine the individual dose is the administration of tracer doses and the kinetic study of each patient. However to develop these measurements it is essential to have reliable operating procedures and calibrated equipment.[3-5]

Nuclear medicine imaging is an excellent tool to provide quantitative information about the distribution of an injected radiopharmaceutical in the body. Acquisition of scans at different preselected times on the same patient provides information on the radiopharmaceutical biokinetics, i.e., the change in the distribution of the radiopharmaceutical in the body over time[6-8].

Generating images from scintillation cameras suitable for quantitative tasks requires additional attention to data acquisition and processing compared to those used solely for qualitative (i.e., visual) interpretation. Moreover, absolute quantitation is more challenging than relative quantitation as it imposes greater demands on the accuracy of corrections for scatter, attenuation, partial volume, and other effects. In addition, there is the need for calibration of the imaging system and activity measurement instruments [9-10].

This paper presents the results of image quantification of Uruguay that participated in a comparison study performed between 9 groups of medical physicists working in institutions in different countries. Our work was done with 133Ba sources calibrated and donated by the National Institute of Standards and Technology-USA (NIST), within the framework of the Coordinated Research Project E2.10.07, “Development of Quantitative Nuclear Medicine Imaging for Patient Specific Dosimetry”. The participants at the various sites each imaged a phantom containing a different set of 133Ba source inserts calibrated by the U.S. National Institute of Standards and Technology (NIST). Images were obtained and activities in the sources were quantified using two different modalities: planar and SPECT.

The main purposes of the project were to investigate the capabilities for quantitative single photon imaging at different sites worldwide, including sites where resources are limited; and develop and test quantitative imaging methods in nuclear medicine practice. The results of all the participants were published in [11], but in this particular paper we describe the results of Uruguay in the comparison work and include data that was not previously used.

2. MATERIALS AND METHODS

In order to quantify nuclear medicine images, we used 133Ba as surrogate of 131I due to long half-life of 10.540 years and similarities between the 133Ba 131I decay schemes [11]. Our centre received 4 sources of 133Ba, epoxy-filled poly(methyl methacrylate) (PMMA) cylinders, each having a length of 3.8 cm and inside diameters of 0.794 cm, 1.27 cm, 1.43 cm, and 2.86 cm to give nominal volumes of 2 mL, 4 mL, 6 mL, 23 mL, respectively. They were measured in Capintec CRC 5 ionization chamber and in a Nucline Spirit DH-V, MEDISO-Hungary gamma camera with high-energy collimator.

Planar and SPECT images were acquired from the sources inserted into a Jaszczak phantom filled with water, as shown in Fig.1. All measurements were made in triplicate. The images were acquired in multiple energy windows to apply the Triple Energy Window (TWE) method for scatter correction. The 4 mL source in air (source 3), was measured to calculate the calibration factor, in cps / MBq in gamma camera using the calibration value, as measured by NIST and also the activities of the other sources. These values were compared with measurements acquired from the image quantification both using Planar end SPECT with ImageJ software.

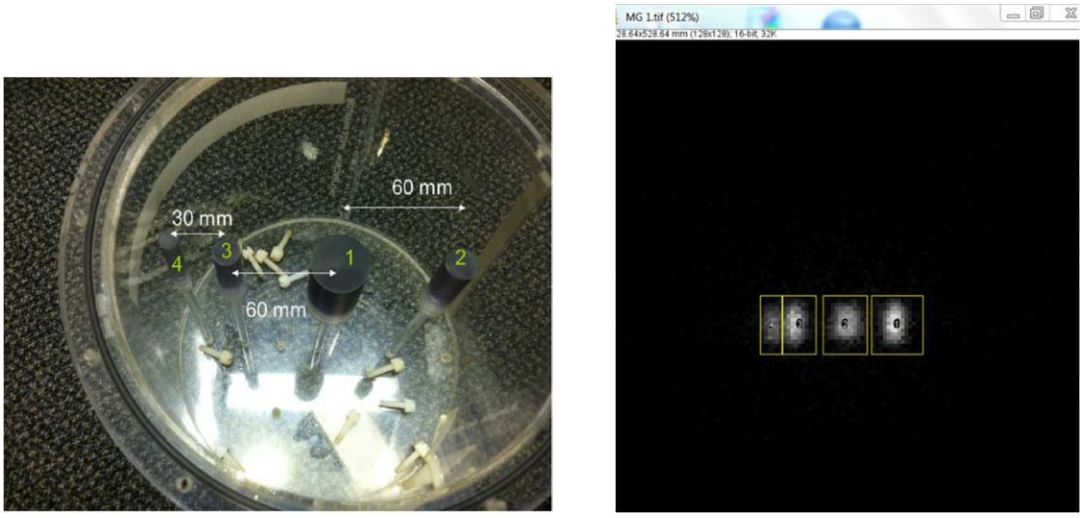
Activities were calculated from the observed count rates and the sensitivity factor using well-established methods and compared with the activities calculated form source . All activities are referred to calibration time. Acquisition and reconstruction parameters are presented in table 1.

TABLE 1.- ACQUISITION AND RECONSTRUCTION PARAMETERS

General parameters			
Collimator	High energy		
Windows	Peak	356 keV	15% window
	Lower scatter	321 keV	5% window
	Upper scatter	403 keV	10% window
Matrix size	128x128		
Acquisition parameters			
Scan duration	600s	60 s per view Step and shoot 120 total views Auto contour enabled	
Reconstruction and post processing			
		OSEM with at least 50 updates No post-reconstruction filter	
Scan set up		Planar	SPECT
Attenuation correction method		Transmission measurements Attenuation coefficient to be used for 133Ba: 0.111 cm−1	Chang-AC
Scatter correction parameters		1.66875 (lower window) 0.6675 (upper window) ROI/VOI to be drawn to include the majority of counts	Triple Energy Window Method

Transmission images for conjugate-view planar activity quantification method were performed with a 67Co flood source. SPECT quantification was done without CT-based attenuation correction, so we used the zero-order Chang correction method.[7]

In the figures 1 and 2 the positioning of the sources in the phantom for planar images is shown.



a) b)

FIG. 1. a)Positioning of the sources in the phantom. Source 1 (23 mL); source 2 (6mL); source 3 (4 mL) (used as reference); source 4 (2 mL). b) ROIs for quantification

3. RESULTS

3.1. Planar studies

Results of planar images quantification are shown in table 2.

TABLE 2.- PLANAR QUANTIFICATION RESULTS

Source number	Reported Mean (n=3)	Reference values (KBq)	Ratio reported /reference
4 smallest	509,6 ± 59	436,6	1.17
3 (reference)	1189,8 ± 0.5	1095,2	1.10
2	1017,6 ± 14	1409,7	0.72
1 largest	1567,3 ± 21	1346,8	1.17

3.2. SPECT studies

Results of SPECT images quantification are shown in table 3.

TABLE 3.- SPECT QUANTIFICATION RESULTS

Source number	Reported Mean (n=3)	Reference values (KBq)	Ratio reported /reference
4 smallest	312,9 ± 29	436,6	0.71
3 (reference)	1185,1 ± 0.1	1095,2	1.08
2	1584,3 ± 60	1409,7	1.12
1 largest	1464,5 ± 41	1346,8	1.09

The ratio was calculated between the mean measured values that were reported in the intercomparison and the reference calibrated values reported by NIST. Despite the limitations associated with the measurements they were quite adjusted to the NIST values.

4. CONCLUSIONS

The results confirm the well-known problems associated with planar imaging, including the unwanted contribution of events from over and underlying activities, self-attenuation in distributed source volumes and the problems of measuring attenuation factors for planar quantitation. These factors are likely to decrease the accuracy and increase the variability of planar quantitation while having a smaller effect on SPECT quantitation.

Nevertheless, these factors are beyond the scope of this work. Thanks to our participation in the exercise we developed new protocols to improve image quantification in therapy. The phantom geometry for planar and SPECT quantification worked quite well, for SPECT studies, the Chang-AC method also performed reasonably well, with *R* values and uncertainties similar to those obtained with planar imaging.

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RADIATION PROTECTION DURING FDG QUALITY CONTROL PROCEDURES IN THE N.N.ALEXANDROV NATIONAL CANCER CENTRE OF BELARUS

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Abstract

The center of positron emission tomography is operating, which is equipped with the cyclotron Cyclone 18/9 IBA for the production of the ^{18}F is based on NCCB. One of the most important stages in the preparation of FDG is the quality control procedure for its physico-chemical properties. The implementation of radiation monitoring and ensuring radiation safety at all stages of the FDG quality control procedure is discussed in the paper.

1. INTRODUCTION

From October 7, 2015, based on NCCB center of positron emission tomography is operating, which includes two main structural units: the cyclotron-radiochemical laboratory and the PET-CT diagnostic laboratory. The cyclotron-radiochemical laboratory is equipped with the cyclotron Cyclone 18/9 IBA for the production of the ^{18}F.
Based on this radioisotope, a 2-fluorodeoxyglucose radiopharmaceutical is synthesized. One of the most important stages in the preparation of FDG is the quality control procedure for its physico-chemical properties. During this procedure, the drug has a high activity, so it is necessary to provide radiation safety when operating the radioactive isotope ^{18}F by laboratory personnel [1].

2. METHODS

To measure the absorbed dose rate (background indices in the room) and the ambient dose equivalent, the dosimeter-radiometers MKS-AT1117M "Atomtech" with the corresponding detection units located at the points predetermined by the calculation of radiation protection and portable dosimeters of "Atomtech" DKS-AT1121 were used [2].

The implementation of radiation monitoring and ensuring radiation safety is provided at all stages of the quality control procedure [3, 4]:

- a container with a radioactive isotope with an average activity of 3 GBq and a volume of 2 ml, enters the quality control laboratory. At this stage, verification of the data of the label with information about the time of production, name, activity, expiration date of the drug in the hood is done. The dose rate is measured directly near the hands, at the chest level and behind the staff while they are working with the open container;
- the vial is placed in a pre-calibrator to measure volumetric activity and further calculation of half-life time value;
- laboratory personnel take several samples from the vial to perform the necessary studies: gamma spectrometry, thin-layer spectrometry (including application of the substance to a special plate), pH test, identification of the main substance and determination of the content of impurities and toxins using gas and liquid chromatographs. Due to direct contact with the drug, the personnel is equipped with dosimeters on the hands and head, which ensures continuous monitoring in accordance with the norms and rules of radiation safety.

During all stages of the synthesis of the radiopharmaceutical, continuous measurements of the radiation background by dosimeter-radiometers MKS-AT1117M, installed in each room involved in production, are carried out according to the radiation safety design. Also, additional dosimetric monitoring is carried out with the portable dosimeter DKS-AT1121, when each subsequent stage of production is performed [2].

3. RESULTS

The period of daily radiation monitoring in the laboratory of quality control of the radiopharmaceutical was 1 month. At present, periodic radiation monitoring is carried out once a week. In the period from 2015 to 2018, based on the results of inspections, there were no exceedances of reference power levels of ambient dose.

4. CONCLUSIONS

Radiation safety of laboratory personnel while they are providing the FDG quality control is constantly monitored, due to the increased likelihood of personnel contamination with liquid radioactive material. The quality control of a radiopharmaceutical requires the accuracy and concentration of personnel, thus it is necessary to minimize the time required to perform this procedure.

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RADIATION SAFETY DESIGN AND ORGANISATION OF THE FIRST CYCLOTRON-PET FACILITY IN ALGERIA

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Abstract

The advantage of setting up the positron emission tomography (PET) integrated systems, with PET/CT unit, cyclotron and radiopharmaceutical laboratory, is the possibility of using the positron emitter radiotracers such as 11 C, 13 N, 18 F, 64 Cu, 124 I, 15 O in the same place as the production. Owing to the progress made in the PET procedures, it is now possible to have not only a highly innovative system of diagnostic examination, with a remarkable improvement in the diagnostic quality and patient care, but also a considerable increase in the number of daily examinations. This paper refers to the acquired know-how with respect to radioprotection in the planning, design, setting up and management of the first Cyclotron-PET facility in Algeria installed at Hôpital Chahids Mahmoudi (TizOuzou, Algeria). The unit is composed of an unshielded 16-MeV energy cyclotron; two radiopharmacy laboratories one CQ laboratory and a PET/CT systems. Specifically, the authors analyze the safety problems connected to the production and the utilization of 18F(FDG), which at the moment is the most largely used radioisotope.

1. INTRODUCTION

The cyclotron is the most widely used particle accelerator for the production of Positron Emission Tomography (PET) radioisotopes (such as 18F, 13N, and 11C). The bombardment of Medical Cyclotron targets with energetic proton or deuteron beam results in the production of intense fields of secondary neutrons and gamma rays causing the enhanced risk of radiological hazards. Unlike high-energy multipurpose research cyclotrons operated by large National Laboratories or Universities, the Medical Cyclotrons are usually located in urban hospitals and frequently visited by members of the public with no radiation awareness training. During the routine operation of a Medical Cyclotron a large number of high energy photons (gamma rays) and fast neutrons are produced by the target as a result of proton induced nuclear reactions with the target material [1]. These by- product radiation fields may leak out through inadequate cyclotron shielding causing significant personal radiation exposure. The activation of cyclotron components [2], production of gaseous radioactive effluent [3] and radiation induced irreversible damage of electronic component of the cyclotron instrumentation and control systems are the major problems associated with the adverse effects of the by-product radiation fields.

The hospital based Medical Cyclotrons are routinely operated by medical technologists to produce large activities of various short-lived medical radioisotopes. Therefore, radiological safety and operational health physics play vital roles in the safe and economic operation of Medical Cyclotron facilities.

The paper highlightes the important aspects of radiological safety and operational health physics of the first Medical Cyclotrons installed in Algeria at Hôpital Chahids Mahmoudi (Tiz Ouzou, Algeria).

2. MATERIAL AND METHODS

2.1. The facility design and structural shielding

The unit is composed of an unshielded GE PETtrace 840, 16-MeV energy cyclotron; two radiopharmacy laboratories and one CQ laboratory (Fig1). The maximum number of cycles per day, number of working days per week, type of radiopharmaceuticals produced, type of procedures performed and radiopharmaceutical activity used are listed in Table 1.

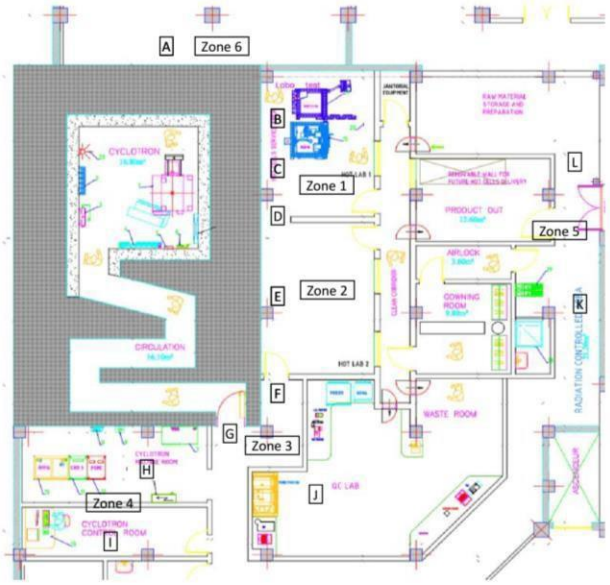

Fig. 1: Cyclotron facility layout

TABLE 1. Operational parameters

Radioisotopes to be produced	^{18}F, ^{11}C, ^{13}N, ^{15}O
Maximal number of cycles per day	1
Maximal duration time per cycle	2 hours
Number of operational days per week	5

The GE PETtrace 840 cyclotron produces two kinds of charged particles protons of 16 MeV (current 130 μA) and deuterons of 8 MeV (current 60 μA), the beam is stopped in the target material and produces secondary particles neutron and radiation gamma rays.

The thickness of the shielding around the cyclotron vault depends on the type of cyclotron, the energy, types of particles, and the targets to be used. The main purpose of the shielding is to reduce the neutron flux during the operation of the machine. Any shielding that will reduce the neutron flux to an acceptable level will also reduce the gamma flux [4].

The primary goal of radiation shielding is to reduce the dose equivalent (DE) at the location of interest outside the shielding imposed by statutory authority, usually 1 mSv/y for public area. The basic geometry of shielding calculation is presented in Fig 2.

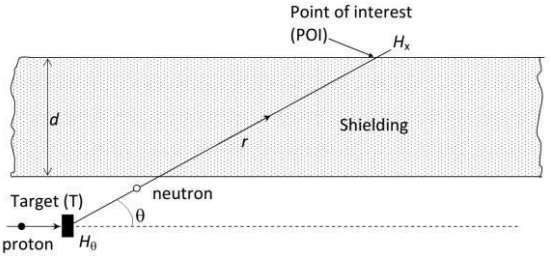

Fig. 2: Geometry for shielding calculation

The (DE) or H_θ at the impact point produced by the protons impinging on the target (T), known as the "source term", reduced to H_x at the point of interest (POI) on the outer surface of the shielding of thickness *d*. The distance between T and POI is designated as *r* [5].

$$H = H_\theta e^{-d/\lambda(E)} \quad (1)$$

Where, λ (E) = energy dependent neutron attenuation length (Figure 5). In principle, H_θ depending on the angle and depth values are necessary for shielding calculation. However, it is often necessary in practice to design shielding in directions transverse (θ = π/2) and longitudinal (θ = 0).

For the neutron and gamma rays shielding calculation, the layouts of the cyclotron vault (Fig 1) were taken into account. The radiation shielding was validated by calculating the neutron and gamma rays (H_x) at the locations shown in Fig 1 under the following considerations:

- A 60 μA current proton beam
- Concrete of a density of 2.35 g.cm3
- Permissible "effective dose" imposed by the statutory authority: 1 mSv/y for public area, and 20 mSv/y for controlled area.
- Neutron and gamma rays (Hx) at the external surface of the shielding walls were calculated using equation (1).

2.2. Real-time radiation surveillance system

A real-time radiation is essential for monitoring the radiation dose levels and other important physical parameters of a medical cyclotron in real-time [6]. A Health Physics RADIABAT HEALTH SYSTEM (Acist Healthcare, France) based on an industrial datalogger has been installed in the facility as shown in Fig 3. The signal outputs (dose rates, liquid levels, stack releases etc.) from the monitoring instruments are sampled every minute by the datalogger, displayed on computer terminal and permanently stored in a CD ROM. For data analysis. The selected signal output from the datalogger are also conencted to the Safety Interlock System.

The RPS subsystem is designed to collect information from the ambient radioactivity measurement probes (GM) and stack. It allows to view, alert and historicize the values collected on the main screen. Installation is also equipped with boxes of monitoring installed in rooms supervised informing stakeholders through columns and displays. The probe values are collected by automata and transmitted to the server.

The VAS subsystem is used to control access in cyclotron vault. VAS controls a round collecting signals of the push buttons installed in the maze. Closing the door of the bunker must intervene in a minimum time. The validation of this round and the closing of the door allow the shooting of the Cyclotron. The VAS system also controls the display of the signals of the cyclotron (BEAM ON / RF ON / MAGNET ON) to inform about the State of the cyclotron.

Radiation surveillance of the working areas was also performed with portable survey meters. Before the procedure for survey was started, the survey meter was checked with a known radiation source and the meter was set at zero in a low-background radiation area. Different steps were performed during the radiation survey

procedure. Radiation survey was carried out in all the locations possible to cover according to the layout. Radiation levels were monitored at the different points indicated in Fig. 1.

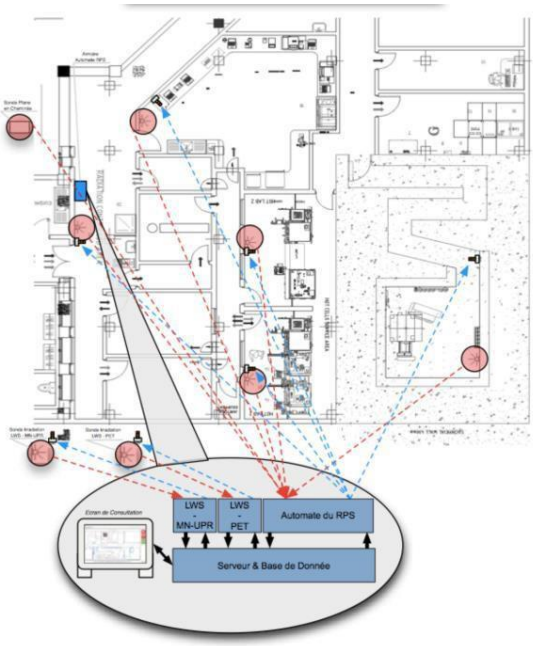

Fig. 3: Implementation plan of the RPS system

2.3. Stack effluent monitoring

Gaseous radioactive effluent in the form of the compounds of ^{18}F (T1/2 = 110 min) and ^{13}N (T1/2 = 13 min) are usually released from the radiochemical processing hot cells of Medical Cyclotrons. Relevant national regulatory authorities often require a mandatory hard copy of the stack release data usually delivered by a real time stack monitoring system [3].

2.4. Operational procedures of safety and radioprotection of workers

The operational management of the cyclotron, radiopharmacy and radiochemistry laboratory involves carrying out a regular preventative maintenance programme and the replacement of components. This programme could cover also radiation exposure.

3. RESULTS AND DISCUSSION

The results of calculation of dose rates (mSv/h) and annual dose (mSv) were performed for different points in the cyclotron facility (Fig. 1). The results are given in Table 2.

These values show that the thickness of the structural shielding of the vault of the cyclotron guarantee an optimum radiological protection and are far from permissible dose to workers since the maximum annual dose is 0.98 mSv to the level of the point A, which is a rarely used technical room. All the calculations were performed for full occupation of areas.

The details of the exposure around cyclotron using RPS system and survey meters are shown in the Table 3. Highest exposure level found in the point A during the bombardment measured with the survey meter (1.02 µSv/h). This is due to the fact that this point is in the direction of the beam. However, the room where this point is located is a technical room where occupation is only occasional. In addition, measurements at the different points of interest lead to annual dose values well below regulatory limits.

.

TABLE 2. Dose rate and annual dose values in the cyclotron facility

POI	Type of area	r (m)	Dose rate (μSv/h)	Annual dose (mSv)
A	Public	4.00	1.31	0.98
B	Controlled	4.00	0.24	0.18
C	Controlled	3.50	0.31	0.24
D	Controlled	4.00	0.24	0.18
E	Controlled	6.50	0.11	0.07
F	Controlled	9.25	0.10	0.03
G	Controlled	9.50	0.10	0.03
H	Controlled	10.25	0.04	0.03
I	Controlled	12.75	0.02	0.02
J	Controlled	12.00	0.03	0.02
K	Monitored	14.00	0.02	0.01
L	Monitored	13.50	0.02	0.02

TABLE 3. The average exposure values with the RPS system and survey meters

POI	Type of area	RPS (μSv/h)	Survey meter (μSv/h)	Measured annual dose (mSv)
A	Public	na	1.02	0.92
B	Controlled	0.22	0.19	0.17
C	Controlled	na	0.20	0.18
D	Controlled	na	0.21	0.19
E	Controlled	0.12	0.13	0.12
F	Controlled	na	0.10	0.09
G	Controlled	na	0.11	0.10
H	Controlled	na	0.10	0.09
I	Controlled	na	0.10	0.09
J	Controlled	-	0.11	0.10
K	Monitored	na	-	-
L	Monitored	na	-	-

TABLE 2. Effective dose for different patient categories

The synthesis of management procedures with related problems of radioprotection is shown in Table 4. T

Operation on the component for its maintenance and other kinds of operation	Operation with exposure	exposure time (minutes)	Frequency of the operation	Kind of protection	Operator
Target service	Replacement of foils and gasket	20	Half-yearly	6 months decay and lead shield	Cyclotron technician
Extraction foil	Replacement	30	Half-yearly		
Target replacement	Unplug	10	Quartely	6 months decay	
Ion source replacement	Removal and housing	60	Half-yearly		
Single delivery Line	Replacement	20	Quartely		
Replacement of all delivery lines	n.a		Yearly		
Radiopharmaceutical quality	Sampling	1	Daily	Short time of operation	Radio-chemist
Control	Preparation For 1 dose	2	Daily		Radiological technician
Dose preparation				Short time of operation	

The operational health physics procedure during cyclotron maintenance is summarised as follows: (a) Determination of the access condition by considering the cyclotron and component cool down time, (b) Radiation dose survey at critical work-areas, (c) Contamination and activation check of cyclotron components, (d) Comply with proper administrative control to prevent contamination and radiation exposure, (e) Well planned waste disposal.

4. CONCLUSIONS

The carried out analysis shows that in-house production and the utilisation of these new radiotracers is becoming a safe. The thicknesses of existing walls provide sufficient protection and meet the standards set by the ICRP 60 (1996). The radiation survey was successfully performed around the cyclotron. All the values are below recommended levels of exposure. This reflected that the exposure levels outside the shield are minimal, which means that there was no leakage of the radiation outside the vault and all the values are below the recommended levels of exposure.

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PROTOTYPE MANUFACTURE OF A RADIOACTIVE BIPHASIC MIXTURE FOR EXTRINSIC CORRECTION EVALUATION MAPS IN SPECT GAMMA CAMERAS

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Abstract

Prototype manufacture of a radioactive biphasic mixture for extrinsic correction evaluation maps in SPECT gamma cameras

Objectives: Propose a unique technique for extrinsic quality control of Variable Manufacturing SPECT Cameras. Methods: an automatic mixer module was prepared to obtain uniform field phantoms, which would allow performing SPECT quality controls. Module consisted of an automatic mixer and a dummy (phantom). Through homogeneity of biphasic mixtures, extrinsic correction maps were obtained, which served also for evaluating equipment quality. It was demonstrated that it depends on mixture homogeneity and optimal distribution of the miscible mixture in the phantom, as well as precise volume quantities, such as: Sodium chloride 0.9% (3.5 lt) and Sodium Pertechnetate Tc 99m (20 mCi), from 0.1 to 0.4 ml. Exposure to radiation was considered preparing the fluid mixture, trying achieve smaller amount of Tc 99m adequately concentrated by volume of distilled water in manikin in shortest possible time. Results: Performance of the analysed mixer module was good and depended of an appropriate design of the mixing system (impeller), conformed by geometry, mathematic model of the mixture, concentration analysis, test container (geometry, volume, diameter), speed(motor power), shape of the mixing blades (angle and number), important components in targets. Conclusion: Efficacy of the biphasic mixture is shown with optimal images acquisition.

1. INTRODUCTION

The present project consists in designing a unique technique for SPECT gamma cameras, recognized by NEMA, IEC. The objective is to reduce time and hand contact while preparing a phantom (manikin) for performing extrinsic quality control of SPECT gamma cameras. There is uncertainty of exposure to the operator and scarce information of absorbed dose in phantom manually filling procedures (1-3).

2. METHODS

A mixture was prepared in plastic container, using a hand instrument, mixing at a distance to hands of 25 and 40 cm from whole body exposure, in 12 minutes. The phantom was delivered to SPECT manually, having a potential irradiation risk with Tc 99m. A prototype of biphasic liquid mixture was built, for evaluating extrinsic correction maps, in SPECT, using automated mixing procedure and phantom filling, free of risk and with needed quality. Design consisted, using ANSYS platform, in building geometry of mesh stirring tank with the aim of not leaving any radioactive waste. Mixing and stirring miscible liquids was done achieving uniform distribution of radioactive component in total water volume, using motor impulse. For accelerating molecular diffusion in liquids used, mechanical energy was used, with a rotational agitator, as has been published (1, 4-5).

2.1. Building the prototype

The base was 1800 rpm motor, to which a stainless steel axis was coupled, with a 17 mm diameter; it had four rectangular cross-shaped blades (shovel), 20 x 24 mm^2. The mesh stirring tank dimensions are: Diameter 25.3 cm, 17 cm height; the cone height was 10 cm. (Fig. 1 and 2)

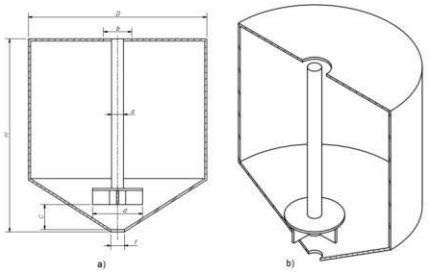

FIG. 1
a) Radial view of agitator
b) Tank isometric view

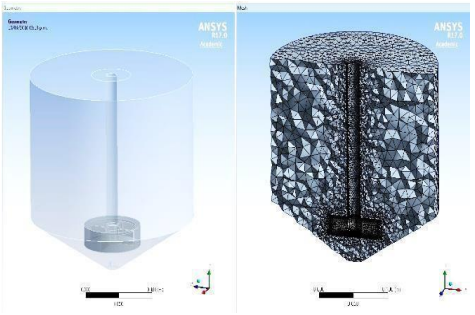

FIG. 2
Mesh stirring tank: Pallet and mesh stirring tank with blades, using the tri-dimensional method for finite element applying the design in ANSYS platform

Equipment for image acquisitions was ECAM SPECT gamma cameras, Siemens, with one and two detectors. Time for mixing and homogenizing the mixture are important variables. Homogeneity does not increase indefinably in time; there is a time for optimal mixture, with a finite interval.

2.2. Experimental evaluation of the built system for preparing the biphasic mixture and automatized phantom filling

A VPA model Veenstra well counter was used, to count micro radioactive drops (cps) from different recipient parts; drops were taken with a 40 microliter Oxford micropipette. Drops were measured using a PTW Freiburg curimeter; for mixture it was used 740 MBq of ^{99m}Tc, 20 samples for each experiment (one tip each).

Experimental procedure

(a) Mesh stirring tank is filled with 3.5 l. of distilled water, to a mark made in external duct, indicating water level.
(b) The duct is connected to the manikin, with the key closed.
(c) Radioactive material to be used in stirring tank is measured.
(d) The motor is on for five seconds, till the on-impulse ceases and movement is stable.
(e) Radioactive material -^{99m}Tc – is added, 30 to 60 seconds are waited, for movement to occur
(f) Motor is turned off, for movement ceasing.
(g) Mesh stirring tank key is opened in extreme connected to phantom, time is registered for material entrance, phantom filling is done for 12 min.; manikin is carefully closed preventing contamination.
(h) In the external visualizer it has to be prevented not having anyradioactive material in the bucket.

A homogenized mixture can be obtained after 25 to 30 seconds. This was verified taking 40 uL micropipette samples and relating positioning of each tip in an essay tube. Samples have been obtained at 5, 10, 15, 20 25, 30, 35, 40 and 45 seconds (Tables 1-2). In each sample there are two measurements: counts per second (cps) and uCi from each tip. Minimal time for automatized mixture is 30 seconds, since in this time liquids are mixed and after, the phantom is filled. Afterwards, extrinsic correction maps are acquired. Qualitatively it was homogeneous and quantitatively was less than 10% for IU and DU in CFOV and UFOV.

2.3. Numeric simulation model of volumetric concentration, velocity and mixture time calculation

Time for mixture is an important variable in the biphasic homogeneous mixture, as well as the volumetric concentration, assuming a constant mixture velocity. Continuity and movement equations of biphasic dynamic fluids are used (6-7)

3. RESULTS

1. **From samples taken at different times and concentration percentage (See tables 1 and 2)**

TABLE 1. CONCENTRATION % OF RADIOACTIVE MATERIAL VERSUS TIME

t_m	Date/hour	Cal. Dec. mCi	uCi	cps e/tip	cps Exper. Value	cps/uCi	Concentration %
5	09/07/2016 14:01	0.000129	0.128914	4769.828571	4.117500	31.939827	0.145665
10	09/07/2016 14:06	0.000128	0.127679	4724.129547	259.015000	2028.639330	9.163215
15	09/07/2016 14:14	0.000126	0.125728	4651.920060	938.385000	7463.637499	33.197397
20	09/07/2016 14:26	0.000123	0.122856	4545.670059	1761.947500	14341.572670	62.332699
25	09/07/2016 14:31	0.000122	0.121679	4502.118664	2143.752500	17618.114580	75.839876
30	09/07/2016 14:42	0.000119	0.119129	4407.768017	2401.667500	20160.248260	84.964176
35	09/07/2016 14:51	0.000117	0.117082	4332.046591	2741.800000	23417.707520	96.997098
40	09/07/2016 15:01	0.000115	0.114850	4249.436205	2826.682500	24612.030270	100.000000
45	09/07/2016 15:10	0.000113	0.112877	4176.432765	2757.932500	24433.172580	97.567820

TABLE 2. ST. DEV. RELATING SAMPLE VERSUS TIME AND VOLUMETRIC CONCENTRATION

t_m	Concentration %	Standard deviation
5	0.145665458	0.70610
10	9.163215182	0.64230
15	33.197396590	0.47240
20	62.332699200	0.26630
25	75.839875900	0.17080
30	84.964176200	0.10630
35	96.997098190	0.02120
40	100.000000000	0.00000
45	97.567820230	0.01720

2. Simulation results

For numerical simulation of fluid volumetric concentration of mixture it has been considered: tank, motor that can be seen in the designed and built prototype. For mixture model – case liquid-liquid -, a turbulent type model numerically resolved, based in finite elements method, was used (7-8). Liquid density and kinematic viscosity were considered. In numerical simulation initial volumetric concentration was taken, as a media of values of experimental analysis, also tank geometry of prototype (cylindrical stirring tank dimensions, with conic base and exit hole 3 cm wide, palette with two blades), as well as time for putting phantom in place, depending on area of transversal section, height and angle section of tank, which is optimally, twelve minutes. Results were performed using available libraries in ANSYS platform, as is shown in following diagrams:

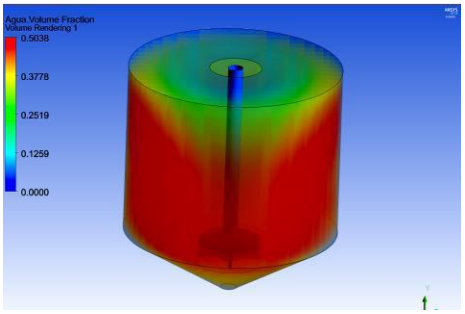

FIG. 3 Development of volumetric concentration of homogeneous mixture and velocity of mixture flux in stirring tank, 3 D view

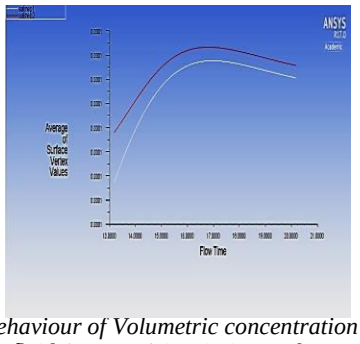

FIG 4 Behaviour of Volumetric concentration of Saline mixture fluid (Tc- 99m) in stirring tank, vs. mixture timing and monitoring steps at different heights Point 1 (white) 0.05 m y Point 2 (red) 0.12 m respectively

4. DISCUSSIONS

Common practice is to deliver the phantom to SPECT equipment manually, having a potential irradiation risk with Tc 99m. An alternative for these controls is Co-57 phantoms, which are not available in Peru nor used due mainly to their higher cost (2, 5). Main results of this project are evaluating efficiency of a prototype for mixing in an automatized manner. The results with the traditional method are similar, in image quality and less than 10% in UI (9). The new method can be considered better because it reduces the mixture time and involves less operator exposition while preparing the mixture. While constructing the new mixture module, it functions using a 1800 rpm motor, and a stirring mixer with conic geometry has been designed and the blades of the automatized system for the mixture and phantom filling and digital mode with a switch on mechanism, facilitates the mixture to be performed in less time, with a homogeneous quality mixture. Transport of the phantom is done without operator exposure, since all the radioactive material is isolated, leading to an extrinsic correction map in 45 seconds of mixture and twelve minutes for phantom charge. In numerical simulation fluid velocity current lines are observed, with a homogeneous mixture, in 20 seconds. These is compared with the experimental data showing that the distribution is uniform in mostly all the tank volume, with velocities ranging from 2 to 6 m/s, very similar to experimental evaluation (10).

5. CONCLUSIONS

The designed prototype of a radioactive biphasic mixture for extrinsic correction evaluation maps in SPECT gamma cameras reduces:

- Radiological risk, which was higher while preparing manually the mixture.
- Time for performing the mixture with the module, without operator exposure in the procedure

These reductions make the prototype a tool with better radiological protection than traditional methods.

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LEAD-FREE PROTECTIVE DRAPE FOR RADIATION EXPOSURE REDUCTION IN POST-YTTRIUM-90 RADIOEMBOLIZATION PATIENTS: A PILOT STUDY

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Abstract

This study was undertaken to explore the potential use of a lead-free protective drape in reducing radiation exposure to the surrounding individuals from post-Yittrium-90 (Y-90) radioembolization patients. Twenty patients (16 males, 4 females, aged 57 ± 8 years) underwent Y-90 radioembolization at the University of Malaya Medical Centre were recruited. All patients had metastatic liver tumours and were treated with Y-90 resin microspheres with administered activity ranged from 0.6 to 3.5 GBq. A custom-made apron consists of lead-free protection drape was worn by the patients after administration of Y-90 microspheres. Radiation exposure from the patients was measured at multiple locations with and without the protection apron. With the use of the protection apron, the external radiation exposure was significantly reduced by 67 ± 17% ($p < 0.001$) at approximately 10 cm away, and 42 ± 24% ($p < 0.001$) at 100 cm away from the patient. The apron is lightweight, lead-free and washable for repeated use. None of the patients complaint about any discomfort or inconveniences due to the use of the apron. The lead-free protection apron significantly reduced external radiation exposure from post-Y-90 radioembolization patients by 30-80%. It is a cost-effective and innovative method to minimize radiation exposure to the personnel and public.

1. INTRODUCTION

Selective internal radiation therapy (SIRT) is one of the minimally invasive therapies currently used to treat unresectable hepatocellular carcinoma (HCC) or liver metastasis. The procedure involves intraarterial administration of radiolabelled microspheres under the guidance of angiography [1]. Currently, only two commercially available radioembolic agents, namely, TheraSphere® and SIR-Spheres®, have been approved by the United States Food and Drug Administration (FDA) for the treatment of liver cancer [2]. The radiolabelled microspheres act as permanent brachytherapy implants localized at the target tumour site until fully decay, delivering maximum radiation from the Yttrium-90 (Y-90) radionuclide *in situ* [3].

Y-90 is a pure beta emitter with high decay energy of 2.28 MeV and physical half-life of 64.2 hours [3]. Although Y-90 microsphere has a mean penetration of 2.5 mm and a maximum penetration of 10 mm in tissues [4], the secondary interactions between beta particles and tissues would lead to the production of

Bremsstrahlung X-ray [5]. The high energetic electron of the beta radiation is deflected in the electric field of an atomic nucleus thus reducing its velocity in the form of kinetic energy and produces Bremsstrahlung X-rays [6]. Bremsstrahlung X-rays can penetrate through patient's body and cause radiation exposure to the medical personnel and relatives surrounding the patient after Y-90 radioembolization treatment. According to the International Commission on Radiological Protection (ICRP) Publication 103, an annual dose limit of 20 mSv is recommended for radiation workers while an annual dose limit of 1 mSv is recommended for members of the public [7]. The excessive radiations generated from the interaction of Y-90 microspheres with tissues would increase radiological burden on medical personnel and relatives of the patients undergoing Y-90 radioembolization treatment.

Unlike the special regulations implemented on patients undergoing Iodine-131 (I-131) thyroid ablation, which involves patient isolation in a shielded ward, there is currently no special regulation being considered for Y-90 radioembolization patient. Generally, decreasing the time of exposure, increasing distance from the radiation source and the use of radiation shielding are three principal methods for radiation protection [8]. While time and distance are sometimes difficult to be manipulated in clinical settings, the use shielding will therefore play an important role in minimising radiation exposure to the personnel.

Recently, a lead-free protective drape is introduced and has been proven to effectively reducing radiation exposure to the personnel during various interventional radiology procedures [9-11]. Therefore, this study aimed to investigate the feasibility and effectiveness of the lead-free protective drape in reducing external radiation exposure to the personnel during post-Y-90 radioembolization treatment.

2. METHODS

Twenty patients undergone Y-90 radioembolization treatment were recruited in this study. Ethical approval was obtained from the Medical Ethics Committee, University of Malaya Medical Centre and written informed consent was obtained from all patients after a brief explanation about the objectives of the study. A custom made apron containing one or two sheets of the lead-free protective drape (Orange coded, 90% reduction at 90 kVp, RadPad® Worldwide Innovations & Technologies Overland Park, Kansas, USA) was designed and worn by the patients after Y-90 radioembolization treatment for a recommended period of 7 days. The design of the apron is shown in FIG. 1. The radiation exposure rate with and without the application of the protective apron was measured using a calibrated survey meter (ASM-990, Victoreen Instrument Co., Cleveland, Ohio, USA) while the patient was at the hospital ward. Each measurement was repeated 3 times and the averaged value was reported. The percentage of radiation exposure reduction was calculated using the following the equation:

$$\text{Percentage of reduction} = \frac{\text{Exposure rate without RadPad-Exposure rate with RadPad}}{\text{Exposureratewithout RadPad}} \text{ x } 100\%$$

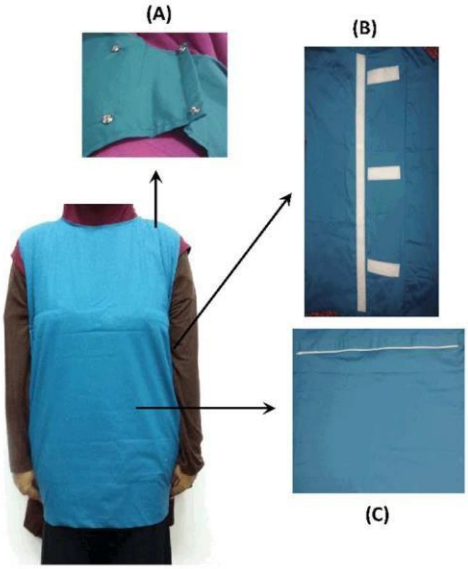

FIG. 1. Design of the custom made, low cost and reusable lead free protective apron. (A) The apron contains some adjustable buttons so that the length of the apron can be adjusted according to the patient's habitus. (B) The sides

of the apron are equipped with Velcro straps so the size of the apron can be adjusted accordingly. (C) There are zipped pockets at the front and back of the apron for insertion of the protective drapes.

3. RESULTS

A total of 20 patients (16 males, 4 females) with the average age of 57 ± 8 years were enrolled into the study. The patients' demographic data and administered activity of the Y-90 microspheres were given in Table 1. Most of the patients were hospitalized for 24 to 48 hours after the insertion of the Y-90 microspheres. The radiation exposure measured at different positions on the abdominal region and at different distances from the patients are shown in FIG. 2. The graph also shows the comparison of radiation reduction when one and two sheets of lead-free protective drapes were used. Generally, a 30–70% reduction in the radiation exposure was observed when one sheet (1.21 mm thickness) of protective drape was applied while 30–80% reduction was observed when two sheets (2.42 mm thickness) of protective drapes were applied.

TABLE 1. DEMOGRAPHIC DATA OF THE PATIENTS.

Patient	Age (year)	Gender	Administered Activity (GBq)	Liver Lobe
1	47	M	1.6	L & R
2	62	M	1.25	L & R
3	50	M	1.2	R
4	50	F	0.8	L
5	49	M	1.5	L & R
6	55	M	3	R
7	56	M	3.5	R
8	54	M	2.6	R
9	55	M	1.6	R
10	60	M	1.5	R
11	62	M	1.5	R
12	73	M	1.5	R
13	48	M	2	R
14	55	M	3.3	L & R
15	50	F	0.6	R
16	65	F	2	L & R
17	49	M	3	R
18	56	F	1.8	L & R
19	72	M	1.5	R
20	74	M	1.5	L & R

M: Male F: Female
L: Left lobe R: Right Lobe

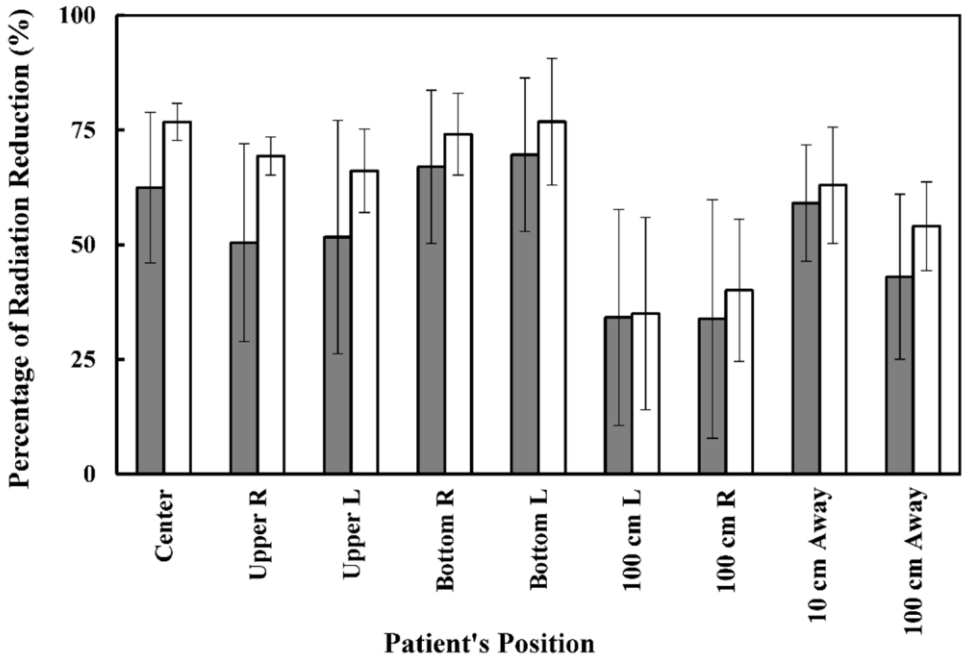

FIG. 2. Percentage of radiation exposure reduction in the absence and presence of 1 (grey bar) and 2 (white bar) sheets of protective drapes at different points on patient abdominal region (central, upper left, upper right, bottom left, bottom right) as well as at 10 cm and 100 cm away from the patient.

4. DISCUSSIONS

The commercially available, sterile, lead-free protective drape used in the study is the orange coded RadPad with 90% attenuation at 90 kVp. The thickness of one RadPad sheet is 1.21 mm. In general, attenuation of X-ray could be due to the absorption or scattering of the X-ray photons when it passes through the material [12] and it is known to be dependent on the X-ray energies, types of material, density and thickness of the material. The RadPad protective drape used in this study is composed of Antimony (Sb-51) and Bismuth (Bi-83). The Bi-83 has an X-ray absorption edge of 90.8 KeV while the Sb-51 is having the X-ray absorption edge of 30.5 KeV. The combination of Bi-83 and Sb-51 in lead free protective drape has increased the X-ray attenuation and has demonstrated its potential in reducing the scattered radiation to medical personnel during various interventional radiology procedures[9-11].

The administered activity of Y-90 microspheres for the 20 patients in this study was within 0.6 to 3.5 GBq, as prescribed by the interventional radiologist. The Y-90 microspheres were administered into either the left lobe (n=1), right lobe (n=12) or both the left and right lobes (n=7) of the patient's liver. A significant reduction of radiation exposure (30 – 80%) was observed when the protective apron was worn on the patient after the Y-90 radioembolization treatment. This shows the great potential of the use of the lead-free protective apron for minimising radiation exposure to the personnel as well as visitors of the patient. In this study, the patients were advised to wear the protective apron for a period of 7 days even after they were discharged from the hospital. Nevertheless, they could remove the apron when they were alone. The 7 days' period was just a suggestion considering the physical half-life of the Y-90 (64.2 h) and the fact that patients were usually scheduled for a follow-up appointment a week after the Y-90 radioembolization treatment, so that the patients could return the protective apron during the follow up session. The optimum period for wearing the apron may need to be assessed further. The apron was designed so that the protective drape could be removed and replaced easily in the apron via the zipped pocket. The apron itself was made of comfortable clothing material and it was washable and reusable to save cost. None of the patients complaint about any discomfort during the period of wearing the apron.

Comparing the thickness of the lead-free protective drape, only marginal difference was found between one (1.21 mm) and two (2.42 mm) RadPad used. This finding was out of our expectation as it did not comply to the exponential attenuation rule. However, further study is needed to determine the optimum thickness of the protective drape. When the RadPad is used in other interventional radiology procedures, it needs to be sterile, single use and non-recyclable. However, in our study, the RadPad was reusable because it was contained inside

the protective apron and it was worn externally. This design would minimize the cost of the apron and make the radiation protection method more affordable.

In conclusion, the lead free protective apron reduced the external radiation exposure from the patients undergoing Y-90 radioembolization treatment by 30-80%. Further studies are needed to investigate the optimum thickness of the protective drape and period of wearing the apron to maximise radiation protection to the personnel and members of the public.

ACKNOWLEDGEMENTS

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CALCULATED DOSE TO FAMILY MEMBERS OF PATIENTS TREATED WITH I131

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Abstract

Patient who receive therapeutic amount of 131I are potentially significant source of radiation to their family members and general public. The aim of the study was to estimate effective dose to family members of patients treated with 131I, released after three days of hospitalization and to compare with dose constraints proposed by international recommendations. The thermoluminiscence dosemeters (TLD) and Radiation Dose Assessment Resource (RADAR) software were used for assessment of effective doses to 60 family members of the same number of hyperthyroid and thyroid cancer patients. Calculated effective doses were well below recommended dose limits except in a few cases. RADAR calculated doses were slightly higher than doses measured by TLD. Hyperthyroid patients should continue to be treated on outpatient basis but they should be well informed for their further behavior to provide minimal radiation hazard for people in their environment. RADAR is very useful software which can be easily implemented in daily routine practice in the field of radiation therapy. It is an appropriate tool for fast estimation of dose to person who might come in close contact with patients who have been treated with 131I.

1. INTRODUCTION

Radioactive iodine 131I has been used for many years to treat benign thyroid disease. Treatment of thyroid cancer with 131INaI is the most common application of radionuclide therapy in nuclear medicine and has been in use for many decades. In the Republic of Macedonia about 50 thyroid cancer patients and the same number if hyperthyroid patientshave been treated with radioiodine 131I every year. These patients present radiation hazard to the other individuals such as hospital staff, the patient's family and members of the public with whom a treated patient may come in close contact. This situation can be overcome by imposing restrictions on the behaviour of the patient, to minimize the dose to close relatives and other individuals. In 1991, the International Commission of Radiological Protection (ICRP) [1] has recommended a radiation constraint of 1 mSv per year to the general population. According to Basic Safety Standards Directive (BSS) [2], the dose limits to the general public are not valid for "exposure of individuals who are knowingly and willingly helping, other than as a part of their occupation, in the support and comfort of inpatient or outpatients undergoing medical diagnosis or treatment". Propose dose constraints from the BSS are 0.3 mSv per episode for the public, 1 mSv for children, 3 mSv for adults up to the age of 60y and 15 mSv for adults older than 60 y. The implementation of this guideline differs among various countries. According to the local hospital rule and old guidelines, the maximum activity given to the hyperthyroid patients treated on outpatient basis is 1110 MBq. The new not yet established, guidelines propose the reduction of the maximum given activity to hyperthyroid patients treated on outpatient basis, from 1110 to 555 MBq. After discharge from the hospital the patient as well as their family members got the brief radiation protection instructions in order to minimize the transfer of radioactive material to the person coming in close contact with patient. There are several papers concerning the subject of doses received by family members of thyroid cancer and hyperthyroid patients [3-8]. Most of the published studies agree that doses to family members are below of the proposed dose constraint of 1 mSv. However, there are several studies that have reported that some children or other persons have received radiation doses higher than proposed dose limit, usually in the case of hyperthyroid patients and their close relatives [9]. In further consideration, the new concepts are taken based on ICRP Publication 103(2007) [10] and BSS and new IAEA BSS (2014) [11]. These concepts take into account the contribution to an appropriate level of protection for people and environment against the detrimental effects of radiation exposure without unduly limiting the desirable human actions that maybe associated with such exposure [10, 11]. The recommendations also update the radiation and tissue weighting factors in the quantities of equivalent and effective dose based on

the latest available scientific information of the biology and physics of radiation exposure. In the ICRP 103 report [10], most tissues have reduction in the tissue weighting factor as compared to the ICRP 60 report [11] and this in turn contributes to the reduction in the effective dose obtained. For example, in ICRP 60 the tissue weighting factors WT for the gonads is 0.20 and the tissue weighting factor recommended by the ICRP 103 is 0.08, but for bone marrow, lungs, breast, colon, stomach, bladder, liver, thyroid and skin they stayed the same or almost the same. Although the new recommendations did not establish changes to the operational quantities and dose limits they could result in the changes of obtained results [1,10,11].

2. MATERIALS AND METHODS

The study population comprised 60 family members of patients treated with radioiodine (30 for hyperthyroidism and 30 for thyroid cancer). The administered dose for treatment of thyroid cancer patients ranged from 3700 MBq to 5550 MBq of 131I. Mean administered dose was 3539 MBq. Twenty-six patients received 3700 MBq, two patients received 4440 MBq and two patients received 5550 MBq of radioiodine. They were hospitalized 3 d after administration and the dose rate was measured every day at four points at the distances of 0.25, 0.5, 1 and 2m. The dose rate measurements were performed with calibrated survey meter type "mini-rad" series 1000, Morgan. According to hospital procedure when the dose rate due to the patient reaches the level of 8 μSv/h at 2 m, patient was discharged from the hospital. Upon the discharge,the patients got radiation safety instructions for their further behaviour in order to minimize the exposure of persons coming in close contact with them, especially children and pregnant woman. Their relatives wore the TLD for 1 week and they were informed not to stay very close to the patient and if so, to reduce the time of staying. It was suggested to maintain the distance between them and patient more than 2m, and not to stay in their vicinity for more than 1 h. All patients who participated in the study gave their consent. Also, family members voluntarily participated in this study realizing the importance and benefits of this investigation. The administered dose for hyperthyroid patients ranged from 185 MBq to 1295 MBq of 131I. Mean administered dose was 683 MBq. External dose rate measurements were performed at the same distance as for thyroid cancer patients, started 15 min after administration of therapy. The patients were interviewed and informed on the research aims by medical physicist and physician. They signed an agreement for receiving a therapy and all patients and their relatives have left positive about the participation in the study. Family members groups were consisted of 12 females and 18 male persons in hyperthyroid group and 24 males and 16 females in the thyroid cancer group. Their age varied from 15 up to 80 y. The effective dose estimations were carried out with thermoluminiscence dosimeters, model TLD 100 which contains hot pressed chips from lithium fluoride (LiF:Mg,Ti) with 3 mm2, encapsulated between two sheets of Teflon 10 mg/cm2 thick and mounted on an aluminium substrate with barcode and within shielded filter holders (type 8814 Harshaw). A detection threshold of a dosimetry system was 0.0054 mSv. TLDs were the most appropriate to estimate radiation dose because the amount of ionizing radiation is directly proportional to the effective dose [12]. Actually, it was estimated Hp [10]. This type of dosimeter has photon energy response for gamma rays in range from 15 keV to 3 MeV (IEC 1066). The TLD reader and cards were calibrated on regular basis. The combined uncertainty of a dosimetry system was less than 15%. The control was kept separately to measure the background. The background readings were subtracted from the readings of estimated effective doses to relatives TLDs. RADAR software is developed [13] on website for rapid electronic access. The program was developed by group of scientists from the Vanderbilt Iniversity, Harbor UCLA Medical Center Nuclear Physics Enterprises. The program is designed to calculate total dose for certain exposure separately for hyperthyroid and thyroid cancer patient for infinite period of time. Besides estimated effective dose might be reached from natural background radiation. Input data are administered activity (in MBq) and average distance (in m) and output is the total dose estimated for given exposure. The calculation was done with values for occupancy factor – 0.25, for extra thyroidal fraction F1 – 0.95 and F2thyroidal fraction was0.05. Teff-1, extra thyroidal, was 0.32 d and Teff-2 thyroid was 7,23 d. These values were used as it was proposed in the program.

3. RESULTS AND DISCUSSION

3.1 Hyperthyroid patients

Effective doses to the relatives of the hyperthyroid patients were estimated using TLDs. The study included 12 females and 18 male relatives who wore TLDs for 1 week. The mean value of the effective dose to relatives of the hyperthyroid patients was 0.87 mSv. The effective dose varied in range from 0.12 to 6.79 mSv. In the three

cases, minimal values were measured at spouses stayed in separated rooms and the most of the time were away from home. Only the spouse of patient marked as number 18 received the highest remarkable dose of 6,79 mSv. The explanation was that the woman did not follow the given recommendations. She stayed very close to her husband all the time after he received the radioiodine therapy. With further analisis, we found that it was woman aged 69 and according to BSS for the adults aged more than 60y the allowed dose constraint is 15 mSv. Eleven family members received effective doses higher than 1 mSv but less than 3 mSv.

3.2 RADAR calculated exposure due to hyperthyroid patients

The estimation of effective doses received by individuals was performed for four distances: 0.25, 0.5, 1 and 2. The results were compared with natural background, All calculations were performed using commercial software RADAR. Table 1 shows the estimated effective dose € received by individual at the distance of 0.25, 0.5, 1 and 2 m from the patient derived from administrated activities and estimated number of days for the same amount of radiation the one received from natural background (BG) radiation according to RADAR calculator.

TABLE 1. RADAR calculated effective doses due to hyperthyroid patient

No.	A(MBq)	E1(mSv)	BG1(d)	E2(mSv)	BG2(d)	E3(mSv)	BG3(d)	E4(mSv)	BG4(d)
1	925	32.0	3885	8.0	971	2.0	243	0.5	61
2	1110	38.4	4613	9.6	1165	2.4	291	0.6	73
3	185	6.4	777	1.6	194	0.4	49	0.1	12
4	370	12.8	1578	3.2	388	0.8	97	0.2	27
5	407	14.1	1700	3.5	427	0.9	107	0.2	27
6	444	15.4	1821	3.8	466	1.0	116	0.2	27
7	555	19.2	2317	4.8	582	1.2	146	0.3	36
8	296	10.2	1214	2.6	311	0.6	78	0.2	27
9	1295	44.8	5463	11.2	1334	2.8	340	0.7	85
Mean	620.8	21.5	2596	5.4	648.7	1.4	163	0.4	41.7

E1, E2, E3, E4 estimated effective dose at distance of 0.25, 0.5, 1.0, 2.0 m respectively.
BG1, BG2, BG3, BG4 equivalence of natural background for 0.25, 0.5, 1.0 and 2.0 m.

3.3 Thyroid cancer patients

Effective doses to relatives of thyroid cancer patients treated with radioiodine 131I were estimated using TL dosemeters. The sum of effective doses varied from 0.02 to 0.51 mSv. The mean value of effective doses to spouses of thyroid cancer patients is 0.21 mSv. Proposed dose constraints for adults are 3 mSv for younger and 15 mSv for older than 60y, respectively, according to the BSSs. All family members were well below the recommended dose constraints, even below the recommended dose for children (1 mSv). The results agree with the previously published paper by Buchan et al. and suggest that the protocol used in performing radioiodine therapy for thyroid cancer patients is done on a safety way [14]. The protocol that includes 3 d of hospitalization and dose rate measurements should be continued. It is recommended to install additional protocol which includes sleeping in separate rooms for 7 days and avoiding of close contact with other people, children or pregnant women. Giving the written instructions on the further behaviour of the patients at home will improve a process of optimization in radiation protection to family members, general public and environment. It was reported that the dose to family members and children of thyroid cancer patients and in all cases the measured doses were lower than 0.5 mSv.

3.4 RADAR- calculated exposure due to Thyroid cancer patients.

The estimation of effective doses received by individuals was also performed for the very same four distances: 0.25, 0.5, 1 and 2 m. Same as in the case of hyperthyroid patients, obtained results were compared with natural background using the very same software and calculator. Table 2 shows the estimated effective dose € received by individual at the four various distances from the patient received cancer treatment doses. Calculated doses by the software were higher than proposed limits, for the activity of 3700 MBq and for distance of 1 m. At a

distance of 2 m, calculated effective dose is 0.3 mSv and estimated dose by TLD was 0.21 mSv. The difference between calculated values and estimated by TLD is higher for the distance of 1 m than for the distance of 2 m. The explanation is that we consider the patient as a [point source but in reality, it is not the situation. According the values of effective doses of family members we might suppose that family members have stayed at the suggested distance of more than 2 m, reduced the time and reduced close contacts with the patient. Even RADAR-calculated doses were overestimated, the software gives opportunity for comparison with the doses that might be reached from natural background. It is easy to implement and perform fast estimation of the effective doses according to given activity before releasing the patient from hospital. The risks not only for the family members but also for neighbours, visitors, co-workers and general public, can be effectively mitigated by the physicist or radiation protection officer with patient specific radiation safety precaution instructions. Current recommendations regarding release of patients after therapy with unsealed radionuclides vary widely around the world. However, the decision to release patient is based on assumption thet the risk can be controlled when the patient returns to their home. [15].

TABLE 2. RADAR-calculated effective dose due to thyroid cancer patients

No.	A(MBq)	E1(mSv)	BG1(d)	E2(mSv)	BG2(d)	E3(mSv)	BG3(d)	E4(mSv)	BG4(d)
1	3700	128	15455	32	3864	8.0	966	2.0	242
2	5550	192	23182	48	5796	12	1449	3.0	362
3	4440	154	18597	39	4709	9.8	1183	2.5	302
Mean	4563.3	158	19078	39.7	4790	9.9	1199	2.5	302

E1, E2, E3, E4 estimated effective dose at distance of 0.25, 0.5, 1.0, 2.0 m respectively.
BG1, BG2, BG3, BG4 equivalence of natural background for 0.25, 0.5, 1.0 and 2.0 m

CONCLUSION

Estimated effective doses were below recommended dose limits except in a few cases. RADAR calculated doses were higher than doses measured using TLD. Hyperthyroid patients should continue to be treated on outpatient basis but they should be well informed for their further behaviour to be sure that they will represent minimal radiation hazard for the people in their environment. After 3 d. of hospitalization, thyroid cancer patient represents minimal radiation hazard to their family members. After release, they should get written instruction on their further behaviour for the next seven days. They should follow advices with the aim to reduce the doses to their close family members and other person that might come in their close proximity. The distance of 2 m from the patient is safe in order proposed dose limits not to be reached. RADAR is very useful software which can be easily implemented in daily routine practices in the field of radiation therapy. It is nan appropriate tool for fast estimation of dose to person who might come in close contact with patients who have been treated with radioiodine.

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TRENDS ON MEDICAL EXPOSURE AND RADIATION PROTECTION IN DIAGNOSTIC NUCLEAR MEDICINE IN RUSSIA

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Abstract

After almost 20 years of stagnation, the retrofit of diagnostic nuclear medicine (NM) units started in Russia. The study is focused on analyzing the trends in the development of NM examinations and effective doses of patients during 2006-2015 period. General trends in the development of NM were evaluated based on the federal statistical data collection forms. More detailed information was obtained by collecting data directly in the NM departments. 3.5 NM exams were performed per 1000 inhabitants in 2015, the average dose per diagnostic examination was 2.5 mSv, and the average per capita dose from NM was 0.008 mSv. Patients effective doses for most common examinations lie in the range of 1 to 4 mSv, the largest average dose, up to 20 mSv corresponds to the examination of lymphatic system with ^{67}Ga-citrate. Doses from internal exposure in PET diagnostics lie in the range of 2-8 mSv. Using equipment combined with CT increases the patient's dose for whole body examination by a factor of 2-3 or more, corresponding to 1-5 mSv for a single anatomical region.

1. INTRODUCTION

Within the last decade a retrofit of nuclear medicine (NM) started in the Russian Federation (RF): old diagnostic equipment was gradually replaced with modern SPECT, SPECT/CT, positron emission tomography (PET) diagnostics using ultra short positron emitting radionuclides (USPER) is rapidly introduced. The structure of diagnostic examination is changing in parallel with the technical re-equipment, the number of high-dose examinations continuously increases, especially when NM equipment is combined with a computer tomograph (SPECT/CT, PET/CT). New questions arise concerning the patient dose assessment, radiation protection of the patients, optimization and justification of the use hybrid examinations, education and training of staff. The actual task in the NM rearrangement period is to monitor the current changes and manage the safety system according to the changing conditions.

The general trends on medical exposure for different types of X-ray and NM examinations can be evaluated using the federal statistical program "The Uniform System for Controlling Individual Doses of the Population" (ESKID), which includes a dedicated form "3-DOZ" and corresponding software for data collection on the patient medical exposure doses. Starting from 1999, this information is summarized in the St-Petersburg Research Institute of Radiation Hygiene (IRH) and annually published in the form of handbooks [1]. ESKID covers virtually all medical facilities in all 85 regions of Russia. However, a significant disadvantage of 3-DOZ form is a lack of verification of the provided data at the first stage of collecting data.

3-DOZ information on NM is focused on the examination structure of the 80-90s of the last century. PET examinations as well as all the hybrid methods (SPECT/CT, PET/CT) are not included into the data collection. No data on the radiopharmaceuticals is included. Hence, it was necessary to assess the actual structure and current trends in NM in Russia as well as levels of patient exposure.

2. MATERIALS AND METHODS

Medical facilities submit data on the number of NM examinations and values of the patients collective effective dose (CED) from these examinations using a 3-DOZ form. This data is artificially divided into "functional" and "scintigraphic" examinations for nine anatomic zones: lungs, heart, skeleton, gastro-intestinal tract, thyroid gland, kidneys, liver and "other" [2]. 3-DOZ forms are transmitted from the medical facilities to the regional

radiation protection authorities (Rospotrebnadzor) where they are summarized together and composed into a regional 3-DOZ form. Regional forms are transmitted to the IRH where the federal database is created

Additionally, there is a federal program of local radiation hygiene certification. Its purpose is to monitor all sources of ionizing radiation operated in the region regardless of their departmental subordination and to assess their impact on the population through the effective dose estimation. The section of medical exposure represents data from organizations subordinated not only to the Ministry of Health, but also to other governmental departments (Ministry of Defense, Ministry of Internal Affairs, Rosatom, etc.). The final result is the "Radiation- Hygienic Passport (RHP) of the Russian Federation", which summarizes information from the regional RHP of all regions of the Russian Federation [3]. The section of medical exposure summarizes information from the regions on the number of diagnostic procedures conducted and on CED for four types of X-ray examinations and nuclear medicine without any other specification.

More specified data on the research types and equipment can be obtained from the federal statistical supervision Form 30 of the Ministry of Health [4]. The combination of these three monitoring systems gives a fairly complete image of the total number of radiology and nuclear medicine diagnostic studies in the country.

A dedicated NM survey was conducted in 2008-2015 by IRH. This survey was based on questioning of the staff and survey of NM departments. A total of 56 NM units from 17 regions of the Russian Federation were surveyed: 42 single-photon diagnostic units, 6 PET centers with cyclotrons, and 8 satellite PET diagnostic units, corresponding to about 30% of all the operating NM units in the country. Data was collected on about 276 thousand NM examinations of adult patients and 12 thousand examinations of children. The largest number of the NM departments was surveyed in St-Petersburg: 18 single-photon diagnostic units, 4 PET centers, 2 PET diagnostic departments; data from 10 single-photon diagnostic units were received in Moscow

The questionnaire included questions about the staff, equipment, types and number of diagnostic examinations per year, the mean values of the radiopharmaceutical administered activity and the estimates of effective doses for patients and personnel. Mean DLP values parameters of CT protocols were included for units working with SPECT/CT and PET/CT. The results of the questionnaire were compared with the data presented in the 3-DOZ form. Additionally, IRH representatives conducted selective inspections of units for checking on-site working conditions and verification of the data presented in thequestionnaire.

The doses of patients were calculated, basing on the data reported by NM departments upon the average introduced activities of radiopharmaceuticals and dose factors from ICRP publications [5] and Russian guideline [6]. When hybrid equipment was used, the dose estimate from CT X-ray was evaluated from the provided DLP values when scanning the "standard" patient (70 +/- 5 kg) [7] and the conversion factors to the effective dose according to the Russian guideline [8]

3. RESULTS AND DISCUSSION

In Russia number of NM units and examinations has been permanently reducing since the 90ths. Only in the second decade of the XXI century the situation has slowed and stabilized. The data from 3-DOZ and RHP forms indicate that the total number of NM diagnostic examinations in Russia has decreased from 630±30 thousand studies in 2006 to the level of 550±30 thousand in 2013-2015, while the patient collective dose of from NM examinations has decreased from 2000 to 1350 Man·Sv [1, 3].

In 2015, 3.5 NM diagnostic procedures per 1000 population were performed on average in Russia, corresponding to the average dose per examination of 2.5 mSv; and dose per capita was estimated as 0.008 mSv. A more detailed analysis of St-Petersburg data indicated that annual number of NM examinations had reached its minimum in 2012 [9]. Their numbers are slightly increasing since then.

About 25 radiopharmaceuticals labeled with ^{99m}Tc, ^{123}I, ^{131}I, ^{67}Ga, ^{111}In, ^{18}F etc. are used in NM in Russia. More than 80% of the examinations are performed with ^{99m}Tc. Radiopharmaceuticals labeled with ^{123}I are mainly used in Moscow and St- Petersburg. In remote regions, ^{131}I-hippurate is used to diagnose kidney diseases. Radionuclide ^{131}I in the form of sodium iodide is used to treat thyroid cancer and hyperthyroidism.

Comparison of the results of surveys in 2008-2009 (15 departments) and 2013-2014 (23 departments) indicated several changes in a structure of NM: reduction of number of renal studies (from 39 to 22%), mainly due to a reduction of renografies with ^{123}I- or ^{131}I-hippurate (from 23% to 5%). The number of thyroid studies also decreased from 18% to 11 % due to a decrease in exams of ^{123}I/^{131}I uptake function in thyroid (from 8% to 1%). At the same time contribution of other examinations is increasing: bones from 22% to 36%, whole body in oncology- from 5% to 13%, brain - from 0.2% to about 3%. Examinations of the lungs, liver, and heart remained at 4-5% each. The current structure of examinations reflects the changes that have occurred in the equipment of the NM units: single-channel spectrometers, scanners and gamma cameras produced in the 80's were replaced by modern SPECT and SPECT/CT. About 20 new PET-centers with cyclotrons for producing positron emitting radionuclides and

satellite diagnostic PET units were installed in 2010-2015. In PET diagnostics, ^{18}F-FDG is mainly used, other radiopharmaceuticals labeled with ^{18}F, ^{11}C, ^{13}N (choline, methionine, ammonium, etc.) are much more rare.

One of the objectives of the conducted survey was to collect the data for the establishment of diagnostic reference levels (DRLs) for the most common examinations, associated with high patient doses. In NM, administered activity to an adult referent patient (weight 70±5 kg) is the most appropriate parameter for establishing DRLs, which is directly connected to the patient dose and is easily measured and monitored by staff. Under the recommendations of international organizations, the range of the controlled parameter between 25% and 75% quantiles of the frequency distribution corresponds to a "good" medical practice.

An example of accumulated data for the examination of the hepatobiliary system with ^{99m}Tc-colloide is presented on Fig. 1. For a sample of 35 hospitals, average administered activity to an adult patient is 130 MBq, 75%-percentile - 150 MBq, and 25%-percentile - 100 MBq. Maximum/minimum ratio of activities is up to a factor of 6. For the majority of the radiopharmaceuticals, the values of the maximum spread of activities among hospitals lies within 2-4.

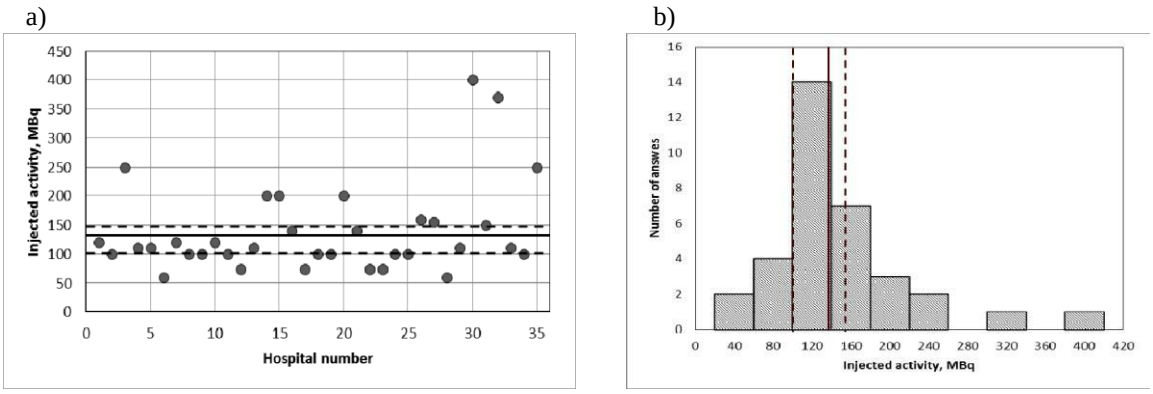

FIG. 1. The average activity of ^{99m}Tc-colloid, administered to patients in different NM units for liver scintigraphy: a) distribution by hospitals (point -value for individual hospital, solid line - mean for all surveyed units, dotted lines 25% and 75% percentiles of distribution); b) the same data as the frequency distribution of the injected activities, the lines indicate the mean and 25% and 75% percentiles.

Mean effective doses for adult reference patients with a body weight of 70±5 kg for the most common diagnostic examinations are presented in Table 1.

TABLE 1. MEAN EFFECTIVE DOSES IN ADULT REFERENCE PATIENTS FOR MOST COMMON DIAGNOSTIC EXAMINATIONS.

Examination	Radiopharmaceutical	Mean	SD	Min	Max
Brain PET/CT	^{18}F-FDG	3.9	1.1	2.4	5.6
WB PET/CT	^{18}F-FDG	17.5	4.3	3.8	23
Brain PET/CT	^{11}C-methionine	4.3	1.2	2.9	6.1
Bone scintigraphy	^{99m}Tc-ph/ph	3.0	0.9	2.1	7.3
Bone SPECT/CT	^{99m}Tc -ph/ph	5.6	1.4	4.3	8.2
Renal scintigraphy	^{99m}Tc -DTPA	0.8	0.5	0.3	2.0
Renal scintigraphy	^{99m}Tc -MAG3	1.1	1.0	0.4	4.1
Myocardial perfusion	^{99m}Tc -MIBI	4.6	0.7	3.2	5.8
Lung perfusion	^{99m}Tc -MAA	1.6	0.6	0.7	3.3
Thyroid scan	^{99m}Tc -pertechnetate	1.3	0.5	0.5	2.6
Liver scintigraphy	^{99m}Tc -IDA	2.0	0.8	1.0	3.4
Liver scintigraphy	^{99m}Tc -colloids	1.3	0.8	0.5	4.2
Lymphatic sistem	^{68}Ga-citrate	20.2	7.8	11.1	30

Effective doses from single-photon scintigraphy for the majority of examinations lie in the range of 1 to 4 mSv. An exception is the scintigraphy of the lymphatic system with ^{67}Ga, with the average doses in several hospitals

reaching 30 mSv, and individual patient doses up to 60-80 mSv. In PET diagnostics, average doses from the radiopharmaceuticals with USPER are about 5 mSv with a range of 2 to 8 mSv. Doses for PET examinations in Table 1 are presented for PET/CT equipment. X-ray exposure from CT scanning gives the most part of the total patient dose.

An important component of data collection on medical exposure is the feedback from the radiologists with an analysis of the data provided by the department. Staff was informed about the errors in effective dose estimation and reporting data to the 3-DOS form. Guidelines on radiation protection in NM were provided for the staff. Subsequent data collection in hospitals indicated the effectiveness of such direct communication.

CONCLUSIONS

Number of NM examinations in the Russian Federation stopped to decrease and stabilized in 2013-2015 at a level of 520-550 thousand examinations per year; in 2015, 3.5 radionuclide examinations per 1000 inhabitants were performed with the corresponding average dose per examination of 2.5 mSv.

Retrofit of diagnostic equipment in NM has led to major changes in the structure of NM diagnostics with an increase in the number of modern scintigraphic examinations replacing radiometric and radiographic procedures. PET diagnostic is rapidly developing: about 25 PET centers and satellite PET diagnostic departments in Russian Federation in 2015, compared to four PET-centers in 2009.

Effective doses from single-photon scintigraphy for most examinations lie in the range of 1 to 4 mSv. The highest doses - about 20 mSv - correspond to the examinations of the lymphatic system with ^{67}Ga-citrate. Scintigraphic studies of heart correspond to an average effective dose of 4-5 mSv. In PET diagnostics, effective doses of internal irradiation of patients are estimated within 2-8 mSv with an average value of 5 mSv. Hybrid examinations (SPECT/CT, PET/CT) increase the patient doses for the whole body examinations by a factor of 2-3 or more, corresponding to 1-5 mSv for one anatomical region .

With the planned commissioning of new PET centers and the modernization of the diagnostic equipment of the existing NM units, an increase in the number of high-dose examinations and an increase in the contribution of NM to the medical exposure of Russian population can be expected.

The accumulation of data continues to establish DRL based on the administered activity and DLP values (for hybrid equipment) for the most common and high-dose examinations.

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